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Edited by
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REACTOR NOISE — SMORN V

Proceedings of the Fifth Specialists
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Munich, F.R.G.,
12–16 October 1987

Edited by
T. D. BEYNON
University of Birmingham, U.K.

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SPECIALISTS MEETING ON
REACTOR NOISE: SMORN — V

Munich, FRG
12th – 16th October, 1987

Hosted by
The European Patent Office

Organized by
GRS with the collaboration of
the IAEA International Working Group on
Nuclear Power Plant Control and Instrumentation
THE OECD (NEA) AND SUB-COMMITTEES

The Organisation for Economic Co-operation and Development (OECD) was set up under a Convention signed in Paris on 14th December, 1960, which provides that the OECD shall promote policies designed:

- to achieve the highest sustainable economic growth and employment and a rising standard of living in Member countries, while maintaining financial stability, and thus to contribute to the development of the world economy;
- to contribute to sound economic expansion in Member as well as non-member countries in the process of economic development;
- to contribute to the expansion of world trade on a multilateral, non-discriminatory basis in accordance with international obligations.

The Members of OECD are Australia, Austria, Belgium, Canada, Denmark, Finland, France, the Federal Republic of Germany, Greece, Iceland, Italy, Japan, Luxembourg, the Netherlands, New Zealand, Norway, Portugal, Spain, Sweden, Switzerland, Turkey, the United Kingdom and the United States.

The OECD Nuclear Energy Agency (NEA) was established on 20th April 1972, replacing OECD's European Nuclear Energy Agency (ENEA) on the adhesion of Japan as a full member.

NEA now groups all the European Member countries of OECD and Australia, Canada, Japan and the United States. The Commission of the European Communities takes part in the work of the Agency.

The primary objectives of NEA are to promote co-operation between its Member governments on the safety and regulatory aspects of nuclear development, and on assessing the future role of nuclear energy as a contributor to economic progress. This is achieved by:

- encouraging harmonisation of governments' regulatory policies and practices in the nuclear field, with particular reference to the safety of nuclear installations, protection of man against ionising radiation and preservation of the environment, radioactive waste management, and nuclear third party liability and insurance;
- keeping under review the technical and economic characteristics of nuclear power growth and of the nuclear fuel cycle, and assessing demand and supply for the different phases of the nuclear fuel cycle and the potential future contribution of nuclear power to overall energy demand;
- developing exchanges of scientific and technical information on nuclear energy, particularly through participation in common services;
- setting up international research and development programmes and undertakings jointly organised and operated by OECD countries.

In these and related tasks, NEA works in close collaboration with the International Atomic Energy Agency in Vienna, with which it has concluded a Co-operation Agreement, as well as with other international organisations in the nuclear field.

The NEA Committee on the Safety of Nuclear Installations (CSNI) is an international committee made up of scientists and engineers who have responsibilities for nuclear safety research and nuclear licensing. The Committee was set up in 1973 to develop and co-ordinate the Nuclear Energy Agency's work in nuclear safety matters, replacing the former Committee on Reactor Safety Technology (CREST) with its more limited scope.
The Committee's purpose is to foster international co-operation in nuclear safety amongst the OECD member countries. This is done essentially by:

(i) exchanging information about progress in safety research and regulatory matters in the different countries, and maintaining banks of specific data; these arrangements are of immediate benefit to the countries concerned;
(ii) setting up working groups or task forces and arranging specialist meetings, in order to implement co-operation on specific subjects, and establishing international projects; the output of the study groups and meetings goes to enrich the data base available to national regulatory authorities and to the scientific community at large. If it reveals substantial gaps in knowledge or differences between national practices, the Committee may recommend that a unified approach be adopted to the problems involved. The aim here is to minimise differences and to achieve an international consensus wherever possible.

The technical areas at present covered by these activities are as follows: particular aspects of safety research relative to water reactors, fast reactors and high-temperature gas-cooled reactors; probabilistic assessment and reliability analysis, especially with regard to rare events; siting research as concerns protection against external impacts; fuel cycle safety research; the safety of nuclear ships; various safety aspects of steel components in nuclear installations; licensing of nuclear installations and a number of specific exchanges of information.

The Committee has set up a sub-Committee on Licensing which examines a variety of nuclear regulatory problems, provides a forum for the free discussion of licensing questions and reviews the regulatory impact of the conclusions reached by CSNI.

The Nuclear Energy Agency Committee on Reactor Physics (NEACRP) is a committee of individually designated experts, with representation — either directly or through regional arrangements — from all interested NEA countries and from the Commission of the European Communities. The IAEA is normally represented at plenary NEACRP meetings by a qualified observer.

The overall task of the Committee is to “review the existing state of knowledge in selected areas of reactor physics of general interest to the nuclear energy programmes of the countries concerned, identify discrepancies and gaps in this knowledge and promote the initiation and co-ordination of programmes of research to fill the gaps”. This task is approached principally through plenary meetings, which are normally held at laboratories where relevant work is carried out, and through specialist meetings on subjects of particular importance.

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SUMMARY OF SMORN V

W. BASTL (F.R.G.)

SMORN V was the fifth international meeting on reactor noise analysis and applications. The first four meetings were held in 1974 in Casscia (Italy); 1977 in Gatlinburg (USA); 1981 in Tokyo (Japan) and 1984 in Dijon (France).

The symposium was sponsored by the OECD NEA Committees on Reactor Physics (NEACRP) and Safety of Nuclear Installations (CSNI) and organized by the Gesellschaft für Reaktorsicherheit (GRS) mbH, with the cooperation of the IAEA International Working Group on Nuclear Power Plant Control and Instrumentation (IWNPPC1).

Noise analysis continues to gain increasing importance for practical application in process, system and component monitoring. The basic idea of noise analysis, to make use of the inherent natural fluctuations of process signals as a new information source, is still a rich soil for exploring new techniques to enhance transparency of complex process and systems behaviour and thus to improve operability and safety.

Therefore the symposium focused on the development of noise analysis, its application and experience with reference to nuclear power plants (i.e. the involved processes, systems and components), for

- determining variables not directly measurable,
- monitoring the operation during normal and abnormal conditions,
- diagnosing the onset of deviations from normal operational behaviour.

The Symposium was attended by 165 participants from 25 countries and three international organizations. From 94 papers sent, in the Organizing Committee selected 56 for oral presentation; 33 papers were presented in four poster sessions. It should be point out that 13 papers on operational experience with noise analysis systems were given by utility representatives from 11 countries.

The outcome of the meeting can be characterized as follows:

- industrial application of vibration monitoring, loose parts monitoring and partly leakage monitoring meanwhile well established and accepted by users.
  - savings by avoiding plant outrages (were given explicitly)
  - further development by means of improved signal interpretation (analysis tools are implemented in on-site systems, so that plant personnel get a much better insight into the information provided by the systems)
  - enhancement of the systems by diagnostic and prognostic modules (explicit information in signal trends and implementation of feature selection modules lead to a more direct information on developing deficiencies),
- application of process monitoring in the very beginning
  - first investigations concerning selected topics like thermohydraulic surveillance of PWRs, detection of local onset of boiling in PWRs, analysis of neutronic fluctuations in graphite moderated power reactors,
- substantial R&D work in new analysis methodologies, which will open up new applications
  - autoregressive modelling as a means to analyse feedback loops
  - displaced spectra techniques for peak identification
  - nonstationary noise analysis methods to better cope with actual process behaviour
  - pattern recognition techniques to improve automatic interpretation.
This means that matured methods and systems are now available for some specific applications. The importance of the practical applications was underlined by the active participation of many utility representatives.

Much more research work is still to be done in the vast area of process monitoring and in the field of diagnostics and prognostics. It is felt that noise analysis can contribute substantially to ongoing qualification and consequently to extension of the lifetime of plants. Further safety related applications imply

- sensor response time monitoring
- signal validation techniques
- boiling detection in PWRs and FBRS
- stability monitoring in BWRs

In this context the still rising importance of noise analysis to solve safety issues has to be pointed out; this is underlined by the presentations given in the "Safety Sessions".

Considering the outcome of the Symposium, it was the unanimous opinion of the participants to continue the SMORN series with another Symposium in due time (say three years). If one wishes to address the main area of future application, an area where most of the research work has to be done, one could think of the term "Symposium on On-line Diagnostics".

With respect to the physical benchmark exercises performed, the successes in anomaly detection have to be pointed out, but further analyses of the characteristics of the different detection methods need to be done.

When taking into account the future research work to be done and the role of noise analysis for safety applications, it is recommended that we maintain the present situation, with CRP and CSNI as the responsible sponsoring bodies. This holds also for IAEA, the International Working Group for Nuclear Power Plant Control and Instrumentation, as a co-sponsor, because non-OECD countries contributed substantially to the high scientific level of the Symposium.

More specifically the following recommendations have been made:

- to consider updating or rewriting of the state of the art report on noise analysis
- to continue physical benchmark exercises on the basis of the already available material
- to decouple the scheduling of the benchmark exercises and the next Symposium
- to continue the SMORN series and to have SMORN VI in the USA, preferably in 1990.
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OPENING SESSION
WELCOMING ADDRESS ON BEHALF OF THE EUROPEAN PATENT OFFICE

P. G. M. Zwartkuis
Vice-President, E.P.O.

Ladies and Gentleman,

it gives me very great pleasure today to welcome you on behalf of the President of the European Patent Office, Mr Paul Baenild, to this our headquarters building for the Fifth Reactor Noise Conference. It is particularly appropriate that this meeting of distinguished technical experts from throughout the world should choose this technically-based institution as its venue. Our daily business, if you like, is the patent grant procedure which involves the protection of inventions covering all areas of technology and the European Patent Office follows both professionally and privately with great interest the rapidly changing face of technology. Patents cover virtually all fields of technology and this includes of course your industry, the nuclear industry. Our European patent number one was in fact granted to the European Atomic Energy Organisation EURATOM.

In our annual report for 1986, the Office chose specifically, following Chernobyl, to deal with the theme of reactor safety and established on the basis of the EPO's technical patent documentation the current state of the art as seen from patent applications. An additional function of the Office is as a storehouse of technical information which we acquire through the information contained in patent applications from throughout the world.

The European Patent Office, with its unique technical documentation facilities, is in a position to report on the current state of the art as revealed by published patent and other technical documents. Our classified documentation system currently consists of more than 21 000 000 documents. In the coming years, this information source will, through the information revolution, become more easily available to researchers.

The European Patent Office, through its work as a patent granting authority for 13 Western European States, undertakes the responsible task of granting patents on which the technologies of the future will be based. We are aware of our obligation both to Europe and to inventors and researchers.

Let me wish you, the participants, a successful conference and continued international cooperation in this important branch of technology. Let me also wish the organisers continued success in their efforts towards international cooperation and harmonisation, themes which are close to this organisation's heart as well.
INTRODUCTORY REMARKS

J. ROSEN

Head of NEA Data Bank, Saclay, France

On behalf of the Director General of the OECD Nuclear Energy Agency, I welcome you to the Fifth International Symposium on Reactor Noise, SMORN V.

We are pleased to be associated with the Gesellschaft für Reaktor-sicherheit and pay tribute to the support of the Bundesminister für Forschung und Technologie in the organisation of this, the fifth in a series of SMORN meetings which began in 1974 at Cassaccia, near Rome, when noise analysis was simply a promising new approach to monitoring in-core events in operating reactors. Although I understand that relatively simple methods were used, the location by acoustic techniques of the leak in the Superphenix sodium storage vat, to be reported during the meeting, is a timely and striking example of the value of this approach, and of its increasing acceptance in practice. On a more formal level, the fact that the Gesellschaft für Reaktorsicherheit should be hosting this Symposium is a further sign of the increasing significance which noise analysis has gained in overall nuclear safety work. The high reputation of the GRS, and their important role in the nuclear regulatory processes of the Federal Republic, needs no comment. SMORN V, like the previous SMORNs, is organised under the aegis of the OECD Nuclear Energy Agency, and in particular of the Committee on the Safety of Nuclear Installations and the Committee on Reactor Physics.

The NEA Committee on the Safety of Nuclear Installations (CSNI) is an international committee made up of scientists and engineers who have responsibilities for nuclear safety research and nuclear licensing. Its purpose is to foster international co-operation in nuclear safety amongst the OECD Member countries. This is done largely through the traditional methods of cooperation, such as information exchanges, establishment of working groups, and organisation of conferences. Some of these arrangements bring direct benefit to Member countries, for example by improving the data base available to national regulatory authorities and to the scientific community at large. Other questions may be taken up by the Committee itself with the aim of achieving an international consensus wherever possible. The traditional approach to co-operation is reinforced by the creation of co-operative (international) research projects, such as the Programme for the Inspection of Steel Components (PISC) and the Loss of Fluid Test (LOFT) project, and by a novel form of collaboration known as the international standard problem exercise, for testing the performance of computer codes, test methods, etc. used in safety assessments. Such exercises are now being conducted in most sectors of the nuclear safety programme.

The Nuclear Energy Agency Committee on Reactor Physics (NEACRP) is a committee of individually designated experts, with representation - either directly or through regional arrangements - from all interested NEA countries and from the Commission of the European Communities. The IAEA is normally represented at plenary NEACRP meetings by a qualified observer.

The Committee keeps under review the state of knowledge in selected areas of reactor physics of interest to the nuclear energy programmes of the NEA Member countries, identifies discrepancies and gaps in this knowledge and promotes the initiation and coordination of programmes of research to fill them. This task is approached principally through annual committee meetings, supplemented by specialist meetings on subjects of particular importance.

The interdisciplinary nature of reactor noise applications is such that the technique is of great interest within the fields of work covered by both committees.
At SMORN IV in Dijon three years ago, it was clear that many of the applications presented had achieved technical maturity, though occasional regrets were to be heard in the final panel discussion about the rather limited number of representatives present from among industrial end users. The present meeting is notable for the high proportion of papers devoted to operational experience, and to presentations of larger-scale industrial systems, and we may now consider that noise analysis has reached the stage of industrial maturity.

SMORN V has all the ingredients of a good meeting, with 180 participants from some thirty countries and international organisations, and 90 papers for presentation. In the perhaps slightly troubled future of the nuclear power industry, one thing is certain: public confidence in nuclear reactor safety will demand the best possible operating procedures and monitoring equipment. Reactor noise analysis has an important part to play.
NATIONAL WELCOMING ADDRESS

K. Krewer (B.M.F.T.)

On behalf of the Minister for Research and Technology of the Federal Republic of Germany I would like to welcome you to the Fifth Symposium on Reactor Noise here in the European Patent Office at Munich. I am glad that the Symposium is sponsored by the OECD, and the organization has been undertaken by GRS in cooperation with the IAEA. I would like to take the opportunity to say a few words on international cooperation in Reactor Safety Research and our attitude towards it.

Following the accident in Chernobyl, the Federal Republic of Germany launched a number of initiatives to enhance international cooperation on reactor safety. Similar efforts were undertaken by other countries, the IAEA, the OECD and the CEC. Although nuclear safety and radiation protection will always remain the responsibility of the individual countries, the response to the Chernobyl accident will further strengthen international cooperation in these areas.

"International cooperation is necessary because nuclear safety is a world issue. A nuclear accident anywhere in the world will indirectly affect nuclear power facilities everywhere."

When this statement was made by Mr. Loewenstein the EPRI at the 8th Reactor Safety Information Meeting of the Japan Atomic Energy Research Institute in December 1980 after the Three Mile Island Accident, nobody could imagine that it would become very graphic 7 years later with the Chernobyl accident. This accident made clear that nuclear accidents physically and psychologically do affect the international community and are not a purely national problem.

The necessity of international cooperation is based primarily on the task of preventing accidents. Furthermore we have to face the fact of limited funds available for Safety Research. Expensive projects often exceed the capabilities of countries even with larger research budgets. For countries with smaller budgets international cooperation is mostly the only possibility to participate in larger research projects and to maintain a high standard of technical know how.

What is important in this context is the worldwide improvement of safety standards. Knowing the worldwide effect of any nuclear accident - even if it is not physically affecting neighbouring countries - it is essential that the safety standards in all countries are at the same high level. A lower level in only one country would be like a weak link in a chain and would jeopardize the nuclear power development in all countries.

But not only the need to carefully use the limited funds for research, and the worldwide impact of any nuclear accident, demonstrate the necessity for close international cooperation. The amount of information can be significantly enlarged by international cooperation. Particularly this will lead to a better mutual understanding and to a consensus about the goals of research activities and the quality of their results. In this manner international cooperation should also include information exchange on operational experience particularly on incidents. This provides a better basis for the reactor operators and responsible authorities to take proper precautions against similar incidents.

The benefits of international cooperation are obvious and agreed upon by almost all countries. Cost reduction for each country participating in international cooperation, prevention of duplication of work and the larger number of available researchers are only a few of the numerous benefits of international cooperation.
Although everyone agrees to the advantages of international cooperation and despite of many presentations emphasizing the necessity of international cooperation, it has to be stated that not every country which is interested in the nuclear energy is participating in the larger international projects. One major reason for this might be that the idea of the investment-benefit relation is predominant also in the performance of such research projects. In case that this idea cannot be dropped in the future, it should be assured that at least the major results are exchanged on a platform which is accessible for all countries. I feel that the cost of the investment is trivial compared to the risk posed by another severe reactor accident.

Another reason often stated for the limitation of international cooperation is the difficult handling of information of proprietary nature. Sometimes this may be a pretext for consideration which assumes information only to be valuable if it is in exclusive possession. Experiences from various multilateral projects as the ICAP or the 2D/3D Project clearly indicate that information of proprietary nature can be adequately handled in international projects.

The most important objective in international cooperation at any time should be the proliferation of nuclear safety.

Looking back at the history of international cooperation I feel that we are on the right way to realizing this goal as being of great benefit for all nuclear power using countries. In the pre-TMI period we had a more loose international cooperation mainly to reduce costs of the very expensive research programs. In the pre-Chernobyl period the need for a much broader international cooperation was realized after the TMI accident and first steps in this direction were undertaken. Now we are in the post-Chernobyl period when the necessity of a close international cooperation has become obvious to everyone, and we should take this seriously. Most of the international projects had been initiated in the pre-Chernobyl period but it should be realized that within the short time since the Chernobyl accident the international cooperation and communications became much tighter and has been significantly intensified. Also this conference is a further step to intensify this communication procedure.

Considering the main safety issues to be achieved, prevention of accidents is still the most important task. In this context improving the man/machine interface and moreover the monitoring and diagnosis tools have always been a major goal of the research efforts sponsored by the BNFT. There is no doubt that noise analysis methods and correlation techniques play a key role in today’s diagnosis systems. Acoustic and vibration monitoring systems are standard systems which allow for early detection of mechanical deficiencies in the plant. Noise analysis became increasingly important for special on-site measurements when thermohydraulic instabilities or boiling characteristics had to be monitored or the validation of signals had to be performed. Along with highly computerized instrumentation systems to be applied in the near future it will certainly be possible to open up further industrial applications, specifically in the area of plant wide diagnosis and maintenance systems. I am quite sure SHORN V is a further important step in the progress of noise analysis methods. I wish you a successful and interesting meeting.
INTRODUCTORY SPEECH

W. BASTL (F.R.G.)

Good morning ladies and gentlemen, this is the place in the program where Professor Birhofer, director of GRS, should have had his opening speech. However, he has an urgent commitment, so he regrets that he cannot be with us this morning.

Ladies and gentleman, it is my great pleasure to welcome you on behalf of GRS and on behalf of the Organising Committee. Let me say some more words about noise analysis, its merits and its prospects.

Since very often we experience some concern about the term 'noise analysis', let me recall - or for our guests explain - the original meaning of this term, a meaning which is still valid. Noise analysis is the art of extracting from the small fluctuations of a measuring signal, which is taken from a process, information about the properties and the behaviour of this process. The basic methods have been correlation and spectral analysis. This opened a new dimension in the information which could be gained from one signal or one sensor. As well as the static information we get in addition dynamic information.

Though noise analysis started off with the application to steady state processes, methodologies have been broadened and improved over the years. In this context I should like to mention techniques like detrending, short term spectral analysis, or pulse analysis. With respect to practical application, noise analysis began with neutron noise investigation of zero power reactor and gradually spread to power reactors, eventually including thermodynamic and structural phenomena. Along with this shift of interest the door has been opened for industrial application. This was due to three basic facts:

1) The inherent power of noise analysis to draw additional information from one and the same sensor fits extremely well into the basic requirement of any instrumentation, which is to avoid overloading of the mechanical systems by sensors; and this requirement is even more important for the hostile environment of the primary circuit of a nuclear power reactor.

2) The potential of noise analysis to achieve information on not directly measurable physical parameters.

3) The high sensitivity of the method or, in other words, its ability to achieve a reasonable signal to noise ratio for apparently extremely small sensor signal.

Today noise analysis methods in nuclear power reactors are used in the following main areas:

- loose parts monitoring
- preferably in the primary coolant circuit and the steam generator
- vibration monitoring
- for instance on the reactor pressure vessel, with its internal structures
- leakage monitoring
- to ensure the integrity of the pressure boundary
- stability monitoring
- like boiling channel instabilities, feedback loop stability
- signal validation and monitoring of sensor health
- a must for extending the use of computerized information systems
- monitoring of thermodynamic phenomena
- for instance two phase flow measurements.
Among these, loose parts and vibration surveillance systems - often addressed also as early failure detection systems - are already standard applications which are required by national rules and guidelines - international standards are on the way. I should like to stress that they belong to the unique category of systems which provide on-line information about the mechanical status of inaccessible structural parts in the reactor plant. This is the main reason why these systems are so essential for the operation and safety of a plant, and why they will get even more important in the future. In this context I should like to mention some actual cases where noise analysis proved its extreme usefulness:

- excessive vibration amplitudes of the reactor shield, with the consequence of mechanical deficiencies
- undue relaxation of hold down springs of the pressure vessel lid
- loose parts in the primary circuit, in various plants at various occasions
- loosened screws at the secondary core support
- excessive primary coolant pump shaft vibration, in some cases followed by a break

In all these cases noise analysis helped to get an early warning of the upcoming deficiency and permitted close follow-up of the further development of the fault, so that plant operation could be continued, or was used for on-line monitoring in order to get an early warning if a similar failure should occur again. The records are even more impressive if we think about cost/benefit figures which have been reported. In 1981 the existing possibility to apply on-line surveillance to the secondary core support of a German PWR saved at the minimum 1 million DM because inspection could be avoided. A similar amount could be saved in the next year because the inspection of the hold down springs need not be done. Summing up the hardware costs, installation costs, maintenance/repair costs and operation costs over six years (which is the time of operation of the system) we get less than 1 million DM. This means a saving of 2 million, for two failure cases against 1 million DM total system costs. Alternatively, we can take an integral figure reported from EDF plants. Comparing the savings gained from eight nuclear power plants equipped with loose parts monitoring systems and operated over 11 years, they obtained $200 \times 10^6$ against $28 \times 10^6$ total equipment costs.

Considering just these few figures, I think noise analysis has already proved its effectiveness. But we are just at the beginning and up to now we have just begun to use the potential of noise analysis. In order to proceed further we have to attack the following goals:

- make the systems even more user friendly and provide the analysis results in a more transparent way
- industrialize or further customize autoregressive models, pattern recognition, multi sensor signal analysis
- integrate noise analysis into other surveillance and monitoring methods
- make use of expert systems as a means for applying sophisticated methods in an industrial environment

It is my strong belief that diagnosis systems and diagnostic tools in on-line and off-line application is one of the main lines we should follow in our future research and development efforts.

Ladies and gentlemen, we have set up a program for SMORN V which should duly reflect status and future trends in research and development, as well as application of noise analysis methods and systems. We had about 110 papers sent in and therefore it was rather difficult to organize them within the available time. As a whole the program is configured rather similarly to the one in Dijon, so you will get acquainted with it rather easily. Again we have a substantial part of the conference devoted to operational experience, and separate sessions on safety related applications. In order to get a good overview of the situation in the various countries we asked for invited papers - and we got an excellent response; I am very grateful to all the contributors. So I do hope we will have a week of interesting discussions and exchange of ideas, and I am convinced SMORN V will be a further substantial milestone in the progress of noise analysis. As you know there is presently a world-wide discussion on accident management of nuclear power plants. The problems to be solved in this field are important, there is no doubt about it, but prevention of incidents and accidents is certainly the task which should remain the No 1 issue of reactor safety. I am confident that the methods we, the noise community, are able to provide can play and will play an essential role in enhancing incident and accident prevention.
Ladies and gentlemen, as the Chairman of the Organizing Committee, I again cordially welcome you to SMORN V. I thank the OECD for giving us the opportunity to organize the meeting. I am grateful to the Federal Minister for Research and Development, especially to Mr Krewer, for all the support. We, the organizers, did our best to provide the framework for a smoothly and effectively running conference, and I should not close before saying that I wish you every opportunity to look beyond the limits of the conference and to experience a little of the charming city of Munich.
OPERATIONAL EXPERIENCE (PART I)

Session chairman: D. N. Fry (U.S.A.)
SUMMARY OF THE SESSION

Baeyens of Laborelec described the noise monitoring programs at seven PWR's in Belgium. Emphasis has been on loose part monitoring, neutron noise diagnostics and vibration monitoring of turbogenerators. The Belgian utilities realize that noise monitoring can help plant operations and are therefore cooperating with the laboratory to develop practical applications of noise methods. In this regard current development is directed toward: automated techniques including artificial intelligence systems; reducing false alarms in loose part monitoring systems, and improved monitoring methods for turbogenerators.

Kunze presented the experiences with a computer-aided noise monitoring system installed at the Greifswald WWER-440 nuclear power station. Emphasis is currently placed on the development of an integrated automation concept that creates noise diagnostics databases and relieves the power station personnel of the routine work associated with noise measurement and analysis. The automation distinguishes between normal and abnormal noise spectra by: selecting proper noise spectra features; applying learning principles during baseline spectra acquisition; establishing patterns for basic and trend data; establishment of tolerance ranges for normal and abnormal plant conditions; and establishing error probabilities to assure minimum false alarms (less than one per month). The system should provide an efficient noise diagnostic database to supply information regarding changes in vibration characteristics due to aging and wear in the Greifswald plant.

Umeda described an on-line noise surveillance system installed at the Tohoku Electric Power Co. ONAGAWA-1 BWR-4 plant to record data from plant start-up through the 3rd fuel cycle. Several anomalies were detected during the 5-year project including: detection of inferior electronic parts in a recirculation pump speed adjustor; incorrectly constructed control devices in a motor generator; incorrect adjustments of a feedwater control system; drift in a feedwater control valve signal due to valve wear; and m i s a d j u s t m e n t of a p r e s s u r e control valve position sensor. Thus, the usefulness of an on-line system for improvement of plant reliability was demonstrated which resulted in the installation of a noise analysis based "Plant Diagnosis Supporting System" in September 1987.

Diagnostic systems installed on Czechoslovakian nuclear power stations were described in a paper by Lott. The objective of these systems are to provide information to plant operators to aid them in optimizing operation and to guide maintenance workers planning and preparing for repair of equipment. The surveillance system is made up of five subsystems for: vibration monitoring, loose-part monitoring, coolant leakage monitoring, acoustic monitoring in free space, and residual life monitoring for steam generators and pressurizer. The system has proved useful in providing information to diagnose impaired quality of glands in main circulating pumps. This diagnosis provided guidance for inspection during scheduled plant shutdowns thus ensuring reliable operation of the pump during the next operating period.

Messinguiral-Bruynooghe presented the results of in-vessel and under-vessel neutron noise measurements made at the Super-Phenix 1200 MWt LMFR from 5% to 90% nominal power during initial plant start. Overall, the flux noise was low although there was observed an interesting resonance in the vicinity of 1 Hz. Unfortunately, the interdependence of plant operating parameters made it difficult to definitively diagnose the cause of the 1 Hz noise. However, a good hypothesis seems to be that the primary pump is producing hydraulic excitation of the fuel assemblies or control rods which in turn causes the 1 Hz signals in the neutron flux. This hypothesis will hopefully be proven when further measurements are made at various operating conditions.
BELGIAN EXPERIENCE IN NOISE MONITORING OF NUCLEAR POWER PLANTS

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ABSTRACT

Seven nuclear power plants are operating now in Belgium. They are all PWR'S either from Westinghouse or Frasmatome type.

Two are two loop reactors, the others are three loop reactors. Neutron noise monitoring was performed since the commissioning of the first unit with ex-core neutron detectors on a regular basis. In-core neutron noise monitoring was started later on. Loose parts monitoring systems were installed later.

The experience gained with these two types of monitoring will be described in this paper, as well as a new turbogenerator vibration monitoring system which is now installed on two of our nuclear power plants.

1. INTRODUCTION

The Belgian nuclear power stations are installed on two sites – Doel and Tihange.

At Doel we have

Doel 1 – 2 started up in 1974-75 – two loop reactors of 400 MWe each.
Doel 3 – started up in 1982 – three loop 900 MWe.
Doel 4 – started up in 1985 – three loop 980 MWe

and at Tihange

Tihange 1 – started up in 1975 – three loop 870 MWe.
Tihange 2 – started up in 1983 – three loop 900 MWe.
Tihange 3 – started up in 1985 – three loop 980 MWe.

The load factor of these units are very good and the overall capacity factor for the nuclear units was 81.8 % in 1985.

Several monitoring programs were set-up since the start-up of the first unit (Doel 1). Among these the noise monitoring techniques were employed on a regular basis for all the units since their first start-up.
The experience and status of these techniques will be summarized in this paper.

2. NEUTRON NOISE

In the field of neutron noise analysis some preliminary tests were performed on the BR3 reactor in the frame of a Euratom contract.
It was then applied on a periodical basis since the first start-up of the first nuclear unit.

Ex-core neutron noise analysis is performed on a monthly basis on all the units and it is a regulatory requirement that a report should be issued on the results at the end of each fuel cycle, in order to be able to take eventual corrective actions during the refuelling period.

With ex-core neutron noise analysis, the main core vibration modes are being monitored. This means:

- the pendular movement of the core barrel and the integrity of the hold down spring.

- Higher order modes such as thermal shield modes and shell modes of the core (envelope).

Some thermo-hydraulic phenomena were also studied by this technique.

For such an analysis, the eight channels are tape recorded (four upper and lower parts of the power ionisation chambers). The recorded data are then analysed by a fourier analyser which provides the normalised auto and cross spectra as well as the coherence functions. This operation is completely controlled by a computer and the spectra are stored on disks for archival. The characteristics of the peaks are also computed.

As said before, this is being made on a monthly basis, but the operator has also an on-line monitoring system which gives the normalised rms value of the different noise components on a banalised recorder. This simple monitoring system is able to isolate the in and out of phase components of the noise for opposite chambers. If fast evolution is detected, Laborelec is called in to make a spectral analysis of the signals.

With this monitoring system it is possible to have an early warning if some structural changes occur.

In-core neutron noise analysis with movable in-core neutron detectors is being performed at least twice per fuel cycle since problems with fuel assemblies were suspected.

With the in-core neutron noise analysis, we are mainly interested in the vibrations of the fuel assemblies and thermo-hydraulics.

The main problems to which neutron noise has contributed to the diagnostics and monitoring in Belgium were:

- Baffle jetting.
- Thimble vibration.
- Nucleate boiling detection.
- Fuel pin vibration.
- Integrity of the hold down spring.
- Primary flow fluctuations.
- Sensor testing.

The main effort now is to automate the neutron noise analysis and incorporate artificial intelligence techniques. The introduction of expert systems is envisaged in order to make the operator as autonomous as possible. The potentialities of the noise techniques are not fully investigated up to now and an effort should be made on sensor testing for instance.

3. LOOSE PARTS MONITORING

First experience with accelerometric instrumentation on the primary loops were obtained at the BR3 and Doel 1 and 2 plants.

At Tihange 1, the installation was modified to obtain an elementary loose parts detection system. The rectified output signal of a general purpose amplifier was used for level detection and alarm. An audio output was also installed.
At the construction of Doel 3 - 4 and Tihange 2 - 3 a Rockwell vibration and loose parts monitoring system was ordered. The accelerometers were installed at the following locations:

- 2 at the bottom of the reactor vessel
- 2 on the reactor vesselhead
- 2 on the hot plenum of each steam generator

These installations perform a dual signal conditioning for vibrations and loose parts. The vibration monitoring is ensured by a low pass filter combined with a rectifier-threshold detector for alarms. For loose parts monitoring the sensor responses are bandpass filtered at the sensor resonance and again a rectifier-threshold detector activates the alarms. An audio output and tape recorder were installed as well as a locator and spectrum analyser on two of the four units.

It turned out that lots of false alarms were issued in operation, due to noisy wiring and poor sensor locations and/or monitoring.

While recently commissioning a Westinghouse DXMINS (digital metal impact monitoring system) at Doel 1 - 2 plants, these problems were properly solved. This system features an advanced digital signal treatment to reduce false alarms, while maintaining a good sensitivity to loose parts impacts. In practice the alarm levels were set at ten times the background noise.

An alarm starts an analog tape recorder during a preprogrammed time. These raw data can be later analysed off-line and compared with base line data which were obtained by extensive calibration. This analysis requires noise experts.

Minor problems were encountered with this installation but operating experience was very encouraging.

During the 15 year experience with loose parts detection only one loose part was reported with certainty. It was detected by the DXMINS system on the secondary side of a steam generator and its estimated mass was about 100 grams. Probably a lot of small loose parts were not detected by the other systems due to a lack of sensitivity.

This leads us to formulate certain remarks about loose parts monitoring systems.

The installation of sensors is very critical. The most probably impact locations for real loose parts should be well studied and the shockwave path should be as short as possible. Mounting studs are to be avoided.

Using mechanical punchers with the proper amount of kinetic energy according to reg. guide 1.133 is not a good choice to calibrate loose parts monitoring systems. One reason may be that they do not produce a pure elastic impact, as it is the case with real loose parts. Pendulum techniques seem more appropriate.

Furthermore, advanced pattern recognition techniques should reduce the rate of false alarms. Already available digital logics represent a good step in this direction and may be further improved.

4. VIBRATION MONITORING OF LARGE ROTATING MACHINERY

A few years troubleshooting vibrational problems with large turbosets leads to the conclusion that most existing monitoring systems fail to properly integrate the vibration vectors at the rotating speed and twice thereof into their alarm logic. Also lacking is the peak-peak vs. fundamental vibration ratio indicating the presence of other frequencies than the fundamental, which may be caused by such serious problems as shaft cracks or a host of unstable self-excited vibrations (e.g. oil or steam whirls). In case the ratio far exceeds one, it is important to obtain a full description of the frequency spectrum to further diagnose the vibration origin. For shaft supported in sleeve bearings, another interesting feature consists of visualising the journal position within the bearing such as described by its clearances.
A real-time on-line system has been developed to add the above capabilities to those already marketed by numerous manufacturers, such as Bode, Nyquist and Campbell-like analyses at variable speeds for successive coastdown and start-up signatures. It has been implemented in 2 fossil-fueled and 2 nuclear plants to monitor the main turbine-generators over 2 years. As a result, about 20 more systems have been ordered since.

The current implementation of the vector monitoring is among the facilities most appreciated by practitioners and, as such, the sole one presented here.

Unbalances are indeed the most common cause for vibrations. They mostly influence the vibration vectors at the running speed (fundamental). In large turbosets, the latter usually undergo significant and yet acceptable changes due to varying thermal and other conditions. Otherwise unexplained changes of sizable amplitudes usually point to a more serious cause such as blade losses, cracks, etc ... Thus an intelligent monitor must first follow a set of operational parameters best describing the above conditions. During a preliminary observation period, it then automatically establishes parameter-dependent confidence zones within which the vectors normally lie for a healthy machine.

Using active and reactive powers as parameters related to the thermal conditions of the turbine and generator in large turbosets, chi-square statistics are built up by the system to produce a finite number of parameter-dependent confidence ellipses, each associated with an operational zone. Care is taken to incorporate the deterministic spreads caused by the parameter variation within each zone. When enough samples are collected within a zone, both the statistics and corresponding confidence ellipses are frozen and a so-called long-term vector monitoring is activated as long as no major change of the machine such as a re-alignment or a re-balancing is performed. A long-term alarm is issued as soon as the vectors leave their confidence ellipses, Fig. 1.

A short-term chi-square statistics is constantly computed for all vectors. It is based on an exponential weighting of the latest vectors and is independent of the operational parameters. If a vibration vector progressively moves, the corresponding short-term confidence ellipse follows the latest vibration vector and always contains it, Fig. 2. In case of a sudden change, this is no longer the case. A short-term alarm is activated. It increments the number of vibration jumps, which is then memorized, Fig. 3.

As is current in vibration level monitoring, peak-peak amplitudes are also compared to pre-established maximum values obtained from such established norms as VDI 2056 or 2059.

For each sensor, a summary of alarms is available on the coloured displays, as already described in the previous figure captions. It is organized as follows: a first row displays the sensor by their inverse video symbols (number + colour codes) as they appear in the multi-sensor vector plots; entries in a second row indicate for each sensor whether it is or has been in peak-peak alarm with a red or green P in inverse video; entries in a third row yield the number of short-term alarms, i.e. how many times the vibration vector has jumped for each sensor. Red numbers point to the sensors with the most recent jumps. No entry means no past short-term alarm. The last row deals with long-term alarms. Any entry means that the corresponding vector lies outside the confidence ellipse that was established the first time the machine was operated in the same thermal condition as the current one. The user can then scan a special data file to retrieve what has happened when the alarm was activated.

Such features greatly enhance the malfunction diagnosis and the prediction of incoming faults. For example, it was possible to detect coupling slip occurring with a 1000 MW 1500 nuclear rpm unit and Newkirk seal rub with a smaller coal-fired turbine. In the future, such a monitoring will be extended to high-merit auxiliary machines such as feed and primary pumps.

5. CONCLUSIONS - FINAL REMARKS

The situation in Belgium in the field of noise monitoring seems satisfactory for the moment. As good relations exist between the operators and the laboratory, frequent measurements are possible. The utilities understand that these techniques can help them and this is the reason why the laboratory has taken a pragmatic point of view towards these techniques. Emphasis is put on trying to get practical results rather than make theoretical research. The trends are to the development of automated techniques and the introduction of artificial intelligence systems, but it will be done in prudent steps.
In the field of loose parts monitoring systems, the effort is put on trying to make the existing systems better in order to get as low false alarms as possible. That means better understanding of the propagation of impacts, the sensors and the detection techniques. In the field of rotating machines vibration monitoring, a very powerful system has been developed and is being installed on a great number of turbosets.

As a conclusion, the vibration monitoring techniques have been well introduced in the Belgian nuclear power plants and are used on a routinely basis playing their role of early warning systems.
METHOD AND EXPERIENCE OF COMPUTER-AIDED SPECTRA CONTROL IN THE GREIFSWALD NUCLEAR POWER STATION

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Abstract:
At the GDR nuclear power plant in Greifswald with 4 VVER-440-type PWR's have been working noise analysis systems since 1975. They consist of measuring lines for incore and outcore neutron flux, acceleration from pressure vessel, main pumps and steam generators and pressure from the primary circuit. For the surveillance of these signals is developed a method for detecting real anomalies with the help of the auto power spectra of the signals. Instead of it the frequency range from 0 to 10 kHz is divided into subranges with respect to known effects of the behaviour of the primary circuit and the reactor. During a learning period the system estimates a mean auto power spectrum and mean values with standard deviations of the rms values in each region. These data are used for the calculation of limits to compare actual spectra with a basic and a trend feature, because after the learning period every APSD, which is classified as a normal APSD, is used for further learning.

1. INTRODUCTION

While efforts in recent years were focused on the wide application of noise diagnostics for analyses in nuclear power stations /1/, on extensive methodical work and on the successful use of first automatic devices in the primary circuit for loose parts detection or mechanical vibration monitoring /2-4/, emphasis in the further development of noise analyzing systems is currently being placed on an integrated automation concept, the creation of noise diagnostics databases and the introduction of systems that relieve the power station personnel of routine work.

Apart from the GDR also Czechoslovakia and Hungary are sponsoring research work to find out procedures and methods of monitoring nuclear power blocks by means of mechanical noise diagnostics.
systems. Major importance for use in nuclear power stations is
attached to algorithms of a high functional reliability, which
in addition can be run quickly and are available at reasonable
cost.

Described in the following are a number of principles that have
proved their worth in the Greifswald nuclear power station for
the development of noise diagnostic methods of control. Particu-
lar attention shall be paid to the process of development of
control methods, i.e. the process technology itself. The prob-
lems to be solved are equivalent to those of pattern recogni-
tion systems so that the part of the control process to be in-
vestigated hereunder can be considered to be a pattern recogni-
tion problem /5/. Even the control procedure can be looked upon
as a pattern recognition problem. Referring to the global spectra
control system, the approach is outlined for a problem of noise
diagnostics in the Greifswald nuclear power station.

2. GLOBAL SPECTRA CONTROL

Apart from the control of special effects such as loose parts
detection, reactor vibration control or leakage detection, it
is desirable in a nuclear power station to be able to make an
assessment of the general condition of a power station unit.
For this purpose, use should be made of all signals (especially
noise signals) generated in the nuclear power station unit. Of
course, the algorithm used cannot be matched to details because
the control system must be designed to detect as many different
effects as possible which are not exactly known in detail or
which need not necessarily be expected to occur at the stage
of developing a method of control. But it is especially the ef-
fects not taken into account so far that the analysis is designed
to detect and indicate.

A way of representation well known to the engineer is the power
spectrum which allows vibrations and frequencies to be clearly
related to each other and is thus preferably used for a general
description of the characteristics of their signals.

Fig. 1 shows the most important of the possible effects in a
WWER-440 unit against their frequency ranges. To be able to
cover as many individual processes as possible it is advisable
to employ a logarithmic division of the frequency axis. On the
other hand, broad ranges (especially 10 to 500 Hz) clearly show
a definite influence of the rotational frequency of the main
circulating pumps and its harmonics on all signal spectra from
the primary circuit of a WWER-440 unit.
Fig. 1: Frequency ranges of processes observed in WER-440

- Mains frequency
- Rotation of main circulating pumps
- Casing vibrations of main pumps
- Structure-born noise of drives and gearings
- Rotational frequencies
- Meshing frequencies
- Natural vibrations of large pipelines
- Reactor pressure vessel vibrations
- Leakage / flow noise
- Loose parts
- Pressure vibrations in 1st circuit
- Control rod vibrations
- Transport effects
- Impact noise of control rods
- Frequency interval division

Frequency range: 0.1 Hz to 100 kHz
This suggests a subdivision of the frequency axis into sections which substantially follow a logarithmic frequency axis but take into account both natural frequencies of the main circulating pumps and mains frequency fluctuations within the usual limits.

This division of the power spectrum into sections is sufficient for the moment from the point of view of engineering interpretation so that the control of the partial effective values of the frequency ranges greatly reduces the amount of data and promises an interpretation-based check.

Those were the ideas underlying the design of a global spectra control algorithm which shall be described in greater detail below. Prior to this, however, the principles of the classification algorithm are to be discussed.

3. PRINCIPLES OF CLASSIFICATION

The classification of noise signals is aimed at subdividing, using the signals obtained in nuclear power stations, the power stations into classes covering the normal state and various failures.

A general spectra control concept is first of all faced with the problem to distinguish between normal and abnormal. This problem has been dealt with by a number of research workers already /5, 6,7/ whose experience has been considered in the development of the principles of classification.

Major principles are:

- Selection of features
  Emphasis should be placed on features that facilitate interpretation and allow a proper distinction of various states. A great deal of methodical work has to be done here. Section 2 illustrates an example of the development of features for general control problems.

- Learning
  As is the case with other pattern recognition methods, a learning phase will mostly be necessary before classification.

A known method is linear learning (e.g. in /6/) which covers a fixed period and proceeds from a no-information condition. At the end of the learning phase the criteria for classification will have been established. Reinstructions can only be made through a complete replacement and a repetition of the learning phase.
Preference should be given to adaptive learning which has hardly been practised so far because of the required higher mathematical knowledge. But in general it has an exponentially falling memory of the classification criteria resulting in distinct advantages for the running speed.

- Patterns
  It is generally accepted practice today to perform signal classification by means of two patterns, a basic pattern and a trend pattern. The basic pattern is the result of the learning phase while the trend pattern is either completely renewed at regular intervals or updated by adaptive learning. It is the purpose of this method to be able to clearly make out both sudden and slow changes.

- Class characteristics
  Class characteristics are used to assess the condition of a plant as normal or abnormal in each classification algorithm. In general, few fixed class characteristics (e.g. vibration limits from standards /8/) are found which can be used right away. Most of them are to be determined in the learning phase. Good experience has been gained with learnt class characteristics in the form of tolerance ranges. Each feature is furnished with tolerance limits that contain the normal operating state with a predetermined error probability (≤ 5%) according to usual statistical tests /9/.

- Reduction of error probability and false alarm rate
  If certain error probabilities are given for the tolerance range of a feature, a number of ways are possible to achieve acceptable error probabilities for a total process described by several features:
  - specify a minimum size for the tolerance range;
  - allow part of the features to exceed the tolerance range;
  - reject and repeat the measurement (/10/).
  The best results were achieved by a combination of all possible ways.

The application of these principles in the Greifswald nuclear power station was subject to two constraints that are met by the control systems of noise diagnostics:

- The results of noise diagnostics are of recommending nature.
  A direct coupling with the accident protection system is not intended.
- The maximum permissible false alarm rate of a noise diagnostic control system is one false alarm per month.
4. SPECTRA CONTROL OF NOISE SIGNALS

4.1. Implementation of Spectra Control

The general global noise signal control in the Greifswald nuclear power station is based on a comparison of the noise signal spectra as these are determined by the computer of the noise analyzing system. Spectra control is implemented as a program module in the computer of the noise analyzing system and makes use of its hardware and software (see Fig. 2).

Autospectra of each noise signal are generated from time increments of 1024 samples and added up according to the frequency interval division in Fig. 1 to form the effective value or partial effective values,

$$
\bar{x}_1 = \sqrt{\frac{U_1}{\sum_{k=1}^{P_{xx_k}}} P_{xx_k}},
$$

(1)

which constitute the features. Each signal is collected twice in general. The Nyquist frequencies amount to 128 Hz and 4096 Hz for acceleration signals and 51.2 Hz for neutron flux signals.

4.2. Control Action

Fig. 3 shows the diagram of the control system. Using the FFT algorithm, auto power spectra are generated one after the other from the signals available on the noise analyzing system, from which, in turn, the partial effective values are calculated. As a rule, the mean values from 16 individual time increments are used for this.

The control system is in its instruction condition at any time. During instruction, mean auto power spectra and mean values of the partial effective values are determined from $N$ ($N=16$) auto power spectra and partial effective values according to

$$
\bar{x} = \frac{1}{N} \sum_{k=1}^{N} x_k,
$$

(2)

and the standard deviations of the partial effective values according to

$$
\bar{s} = \sqrt{\frac{1}{N-1} \sum_{k=1}^{N} (x_k - \bar{x})^2}.
$$

(3)

The mean auto power spectrum and the mean values and standard deviations of the partial effective values are both basic and
Fig. 2: Hardware of noise signal spectra control (diagram)
trend pattern at this time. They are stored as a file on the disk store of the noise analyzing computer together with signal identification data. Instruction takes about 30 minutes for one noise signal.

If the instruction of a signal has been completed the signal is subjected to classification in a cyclic manner for which an auto power spectrum and the partial effective values are determined again. Classification is performed in two stages: classification proper and trend instruction.

Firstly, the basic and trend patterns of the signal are read in from the disk store and limits are determined for the partial effective values as follows:

\[
1_L = \begin{cases} 
\bar{x} - 4 \sigma & \text{if } 4 \sigma > 0.05 \bar{x} \\
0.95 \bar{x} & \text{if } 4 \sigma \leq 0.05 \bar{x}
\end{cases}
1_U = \begin{cases} 
\bar{x} + 4 \sigma & \text{if } 4 \sigma > 0.05 \bar{x} \\
1.05 \bar{x} & \text{if } 4 \sigma \leq 0.05 \bar{x}
\end{cases}
\]

The range \(1_L\) to \(1_U\) is the tolerance range of the respective effective value. If \(7/8\) of the partial effective values of a signal are within their tolerance ranges the signal is considered to be normal, otherwise the measurement is repeated once, and if the test of the basic and trend patterns is negative again this will lead to the indication of 'abnormal'. Signals classified as 'normal' are subjected to trend instruction, with mean values being re-instructed according to

\[
\bar{x}_{\text{new}} = \frac{N-1}{N} \bar{x}_{\text{old}} + \frac{1}{N} x
\]

and standard deviations according to

\[
\sigma_{\text{new}} = \sqrt{\frac{N-2}{N-1} \sigma_{\text{old}}^2 + \frac{1}{N} (\bar{x}_{\text{old}} - x)^2}
\]

The result of classification is logged, and the new trend pattern replaces the preceding one on the disk store.

4.3. Characteristics of Control Procedure

To be able to assess the results of classification in the correct manner a good knowledge is required of the characteristics of the control procedure. The previous section will be better understood if it is known how often the specified tolerance range is exceeded.

In the nuclear power station, upper deviations from the tolerance range were observed with probabilities of \(p=1.6\) to \(5.5\% (\beta \approx 3.8\%)\). This appears to be representative of the partial effective values of noise signals produced by nuclear power stations. If one wanted
Fig. 3: Flow diagram of noise signal control
to take care of each upper deviation, e.g. with ten partial effective values per signal, this would amount to an error probability of

\[ P = 1 - (1 - p)^{10} \approx 32\% \]  

(7)

so that each third measurement would have to be assessed by an expert. If, on the contrary, single upper deviations of the tolerance range are permitted this will lead to a distinct reduction of the false alarm rate.

In this way, the toleration of 12.5 % upper deviations at \( p = 3.8\% \) will give an error probability of

\[ P = \sum_{i=0}^{n} \binom{n}{i} p^i (1 - p)^{n-i} \approx 0.8\% \]  

(8)

with \( n \approx m/8 \). In fact, the value observed is \( P = 1.7\% \) which reduces the false alarm rate to a reasonable level, i.e. \( P \approx 10^{-4} \) for a single repetition in case of deviation.

Of interest is the functioning of the trend instruction of mean values and standard deviations according to (5) and (6). Both equations have been derived from (2) and (3) by assuming a recursive calculation of the mean value and standard deviation. Using the discrete Laplace transformation /11/ known from digital filtering, the transfer functions of the two equations can be determined taking into account a fixed \( N \). According to this, the trend instruction of the mean value has the behaviour of a simple low-pass filter with a time constant of \( (N-1)\cdot T \), and that of the standard deviation has the behaviour of an effective value generator comprising a simple high-pass filter with a time constant of \( N\cdot T \), a square function converter, a simple low-pass with a time constant of \( 2\cdot(N-1)\cdot T \), and a square-root extracting element.

The behaviour discussed here is shown very well by Fig. 4

Practically, this means that the mean value and the standard deviation complement each other in the description of the behaviour of features in that the mean value with its low-pass effect will detect slow variations while the standard deviation with the input high-pass registers fast feature variations.

4.4. Applications

Fig. 5 shows an example of a comparison of an actual auto spectrum (spectrum under test) with the basic and trend patterns inclusive of logging. The auto spectrum is that of a pres-
Fig. 4: Response of mean value and standard deviation to a unit-step function (top), to a Gauss process ($\sigma = 5\%$) disturbed unit-step function (centre); transfer functions of the effective digital filters (bottom)
Fig. 5: Comparison of patterns and sample for the spectrum of the acceleration signal B303 of the reactor pressure vessel: below: log of check of April 2, 1986, 12.47 p.m.
sure vessel acceleration signal which was classified as 'normal'. The noise analyzing computer will in such a case only print out in this particular case that the signal B303 was checked in April 2, 1986, 12.47 p.m. and that the basic classification showed an upper deviation of the tolerance ranges in the frequency intervals 3 and 11, and the trend classification a lower deviation in the frequency interval 2. (Frequency interval 1 covers the range from 0 Hz to Nyquist frequency.)

The following two examples (Figs. 6 and 7) show abnormal states that have been detected. Fig. 6 illustrates the auto spectrum of a pump signal, compared with its basic pattern. Deviations are quite obvious, in particular the peak at the 6-fold rotational frequency and sidebands with the distance of half of the rotational frequency and the much higher effective value. During the inspection of the pump the rotor was found to have a contact mark.

Similarly, Fig. 7 shows the acceleration signal spectra of a valve. In the frequency range above 2.8 kHz the valve produced a whistling noise which is due to a flow-induced vibration, a typical indication of the increasing wear of this type of valves.

5. SUMMARY

Principles of the development of noise signal control systems were drawn up and described by reference to the application of the global spectra control concept. A way of automating the noise analysis in a nuclear power station was described. The control method described has been used in the Greifswald nuclear power station since 1982 for the continuous control of the noise signals of the station, and for the development of a card register which enables an assessment of the systems' behaviour over years with the aid of their noise signals and which can supply information on changes of vibration characteristics due to ageing and wear.

The card register comprises:
- images of spectra of signals,
- related lists of spectral values,
- mean values and standard deviations of partial effective values,
- trend representations of effective values.

It is intended in the immediate future to continue this work on the noise analyzing systems of the Greifswald nuclear power station, to arrive at a higher degree of automation of the control algorithms and to transfer them to new fields of application.
Fig. 6: Basic pattern and sample of a pump acceleration signal
Fig. 7: Basic and trend patterns of a valve acceleration signal with whistling noise
Emphasis is placed on the creation of an efficient noise diagnostics database.

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EXPERIENCE OF ON-LINE SURVEILLANCE AT ONAGAWA-1 BWR PLANT

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Abstract - An on-line system was installed on ONAGAWA-1 to monitor plant situations and to accumulate basic data for about 4 years, from start up test to 3rd cycle operation. Some kinds of anomalies were undergone during that period. Anomalies can be classified into several categories by growing modes of phenomena. It was verified that the system is useful to identify anomaly cause for the various kinds of anomaly modes. Some examples of experiences are introduced and discussed here.

1. INTRODUCTION
ONAGAWA-1 (Tohoku Electric Power Co., Inc.) is a BWR-4 nuclear power station producing 524 MW electric power, which started commercial operation in June 1984. To attain high reliability and availability for ONAGAWA-1, Tohoku Electric Power Co., Inc. and Toshiba Corp. started a research & development project on plant surveillance and diagnosis before the start up test.

Main purposes of this project are:

(1) To acquire baseline data and analyze the characteristics of the plant using noise analysis technique.
(2) To develop a new diagnosis technique for transient data, including small step-wise or rectangle-wise change (Umeda, 1987).

It was proved that the noise analysis technique is very useful for plant diagnosis, through the application to anomaly inspection on actual plant. And a practical method for transient data diagnosis was developed and verified. In this paper, the topics are concentrated on some experiences gained from diagnosis technique application to actual countermeasure on ONAGAWA-1.

2. BRIEF DESCRIPTION OF THE PROJECT

2.1 Total schedule

Figure-1 shows the total schedule for this project. First of all, process and control signals used for surveillance were selected. Design and manufacturing the on-line surveillance system was accomplished, according to the selection of monitoring signals before going into start-up test. The system began to work with start-up test and there were many chances to collect data for various kinds of operating conditions which occurred during start-up tests (Kishi, 1986).

![Fig. 1 Project schedule](image)
Figure-2 shows the outline to carry out this project, which is divided into two categories of activities. The first activity was to conduct continuous monitoring using an on-line system which was installed in the control room at ONAGAWA-1. This activity aimed to contribute to plant reliability directly on-site by detecting an anomaly as soon as possible. The second activity was off-line data analysis and diagnosis. Raw data were recorded periodically on digital magnetic tape using the on-line system and were sent to laboratory. These data were precisely analyzed and accumulated on data base to detect any change and trend by comparing with historical data.

2.2 Surveillance objects

Surveillance objects should be selected considering many factors, such as trouble effect for plant operation and statistics of troubles etc., because the number of monitoring signals is limited. As a result, signals from main control systems (recirculation flow control, feed-water flow control and pressure control), In-core neutron flux and jet pump flow were selected. A total of 133 signals were branched from sensor transducers and check points in control devices. Surveillance objects are briefly shown in Fig. 3.

2.3 On-line surveillance system

The on-line system continuously samples 64 signals every 100 msec. 32 signals from all sampled data are selected to be used for continuous monitoring. If one of these signals exceeds a specific threshold level, the system starts to record all 64 signals from 60 seconds before trigger to 270 seconds after trigger. This kind of function is very useful for isolating troubles, like abrupt change or unexpected scram, because repeatless abnormal phenomena are very hard to ever catch again. The system has a function of frequency analysis using FFT (APSD, CPSD, Coherency etc.). It is useful to investigate operating characteristics for control devices. Another function of this system is to record data periodically (once every two days) on magnetic tape. This magnetic tape is sent to the laboratory when it becomes full. These data are accumulated to make database and were used to survey long term trends. The system is mainly composed of two micro-computers, digital data recorder, hard disk (20MB), printer-plotter and operator console. Figure 4 shows the configuration of the on-line system.
3. EXPERIENCES IN AN ACTUAL PLANT

3.1 Some examples of experiences

The database such as noise pattern characteristics and long term trend of signals at ONAGAWA-1 was accumulated through the 5 years project, and was used for actual anomaly inspection. The on-line system detected some kinds of anomaly data during start-up test because many devices were on the way of adjustment. After going into the commercial operation of ONAGAWA-1, abnormal situation hardly occurred apparently. However, the symptoms of device characteristics change could be traced through the data analysis of this project. In this section, some examples and the effects of diagnosis for each case are introduced.

CASE-1 Anomaly in recirculation pump speed adjuster

This trouble was automatically detected by the on-line monitoring system, during the 3rd cycle of plant operation. Figure 5 shows the raw data recorded by automatic trigger. It was found that the phenomenon was a abrupt single-shot change and was settled after a few seconds. The trouble seemed to have happened in recirculation flow control system. Disturbance amplitude was so small that the effect did not propagate to main process, that is, recirculation flow rate and jet pump flow rate etc. So it might be difficult for operators to be aware of this symptom by watching recorder charts in the control room. After detecting this phenomenon, inspection of suspected devices was done quickly, based on recorded data. As a results of quick inspection, inferior quality of electronic parts in a recirculation pump speed adjuster was found to be the cause of that phenomenon. Figure 6 shows the block-diagram of the location where abnormal parts were found. If it were left in that condition, a sudden big transient might occur as a result of development of parts degradation. Automatic anomaly detection showed the ability to catch a symptom before propagating to the overall plant.

CASE-2 Anomaly in PLR MG set generator speed signal

The on-line system has a function to analyze the frequency spectrum for each signal. Anomaly symptom can be detected by observation of spectral pattern change. In the initial stage of the start up test, spectral pattern for the MG set generator speed signal showed a change, compared with the usual case (Fig. 7). Spike noise was observed in raw data (Fig. 8).

![Fig. 5 Automatically recorded raw data by abrupt change monitoring](image)

![Fig. 6 Location of anomaly in recirculation flow control system](image)
This caused spectral level increase in the high frequency region. As a result of field inspection based on recorded data, it was found that control devices were incorrectly constructed, so the control signal was easily affected by disturbance noise. As the control system did not respond against these rapid spike noise, disturbance did not propagate into process signals. If it were left in that condition, a big process transient might occur as a result of disturbance noise. This is the same kind of examples as CASE-1, detected before plant operation was affected.
CASE-3 Feedwater control system hunting

A example of this case is the same as CASE-2, which shows an effect of spectral analysis. Feedwater flow hunting was observed in the process of feedwater control system adjustment in start up test operation. This hunting phenomenon affected the core inlet temperature and neutron flux showed an oscillatory aspect. However, this was settled when adjustment was completed. Power spectral densities for feedwater flow and neutrons flux, before and after control system adjustment, are shown in Fig. 9.

![Fig. 9 Comparison of power spectral densities before and after adjustment for feedwater flow and neutron flux](image)

Figure 10 shows coherence between feedwater flow and neutron flux. It is noticed that coherence is very large at 0.02 Hz before control system adjustment. This shows strong coupling between feedwater flow and neutron flux, due to feedwater flow hunting. Hunting amplitude is largely decreased after adjustment. Then, coupling between feedwater flow and neutron flux becomes weak and coherence decreased to a small value. As stated here, spectral analysis was used to confirm the effect of control system adjustment. It was verified to be very useful for estimating operation characteristics.

![Fig. 10 Comparison of coherence before and after adjustment between feedwater flow and neutron flux](image)
CASE-4 Drift in feedwater control valve signal

The on-line system recorded data periodically. These data were sent to laboratory and accumulated. Long term trends for signals can be watched by the database. This case introduces an example where in feedwater control valve signal drift can be pointed out by the trace of data at first cycle operation all the way.

Trend data are shown in Fig. 11. As noticed in Fig. 11, feedwater flow was kept constant, in spite of control valve signal drift. This fact suggests a change in valve characteristics. The authors recommended precise inspection of valve in the period of annual maintenance. The reason of control valve slight drift is that feedwater control valve position compensates for the excess in feedwater flow by valve wear to keep the flow constant. As this characteristics change progresses bit by bit, it is very hard to isolate a trend without this kind of long term database. This is a typical case that demonstrated the large effect of preventive maintenance.

![Fig. 11](image)

**Fig. 11** Drift detection of feedwater control valve signal

CASE-5 Scram caused by misadjustment of pressure control valve position sensor

Reactor trip occurred just after connecting the main generator to the load at start-up of second cycle operation. The on-line system automatically recorded the data before and after this event. These useful data contributed to searching for the cause of reactor trip. The raw data, recorded at that time, is shown in Fig. 12. The data suggests that one of the four pressure control valve position signals shows a different behavior from the others. The misadjustment of No.1 pressure control valve position sensor turned out as a result of field inspection based on these data. This is a typical case of corrective maintenance, because reactor trip occurred without sensible symptom. However recorded data largely contributed to quickly selecting the anomaly candidates and enabled rapid countermeasures without loss of time.

![Fig. 12](image)

**Fig. 12** Recorded raw data largely contributed to field inspection for countermeasure
3.2 Evaluation of experiences

The examples introduced in the previous section are some anomaly cases encountered in research & development project. These experiences show that the causes of troubles and progress modes are various or repeatless. It is important for efficient surveillance to classify the phenomena into several kinds of mode and watch the condition by using an adequate method of surveillance.

The phenomena were classified into the following three categories based on monitoring methods.

1. Abrupt change monitoring
   Monitoring anomalies where continuous symptoms are not sensible in signals and where repeatless abrupt change in significant magnitude occurs.

2. Frequency analysis monitoring
   Monitoring anomalies where continuous symptoms appear in signals.
   Symptoms are detected by frequency analysis.

3. Long term trend monitoring
   Monitoring the trend of signal characteristics for a long term ranging from several months to one cycle of operation.

Phenomena encountered in this project are categorized as shown in Fig. 13, following the above classification. Figure 13 is expressed in the rate of encountered event number. "Others" means reactor trip case introduced in CASE-5. Figure 13 explains that almost all the anomaly modes can be detected by monitoring methods categorized by the above viewpoint. And it shows that the contribution of frequency analysis monitoring to trouble shooting is only about one forth of total number. According to our results, it should be noted that frequency analysis technique is not all-around for plant surveillance, although this result is not general but is peculiar to ONAGAWA-1 and is limited to initial stage of plant operation. Of course, frequency analysis is very powerful and indispensable to plant diagnosis, but wide spread viewpoint is important for the purpose of plant surveillance.

4. CONCLUDING REMARKS

Following results were obtained through the project.

1. Satisfactory contribution to ONAGAWA-1 reliability improvement could be obtained through start-up test and commercial operation.

2. Usefulness of the on-line system for early anomaly detection is demonstrated.

Based on these results, the Tohoku Electric Power Co., Inc. decided to install a full-scale "Plant Diagnosis Supporting System" using mainly noise analysis technique. The system has been in practical operation since September 1987.

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SYSTMS OF OPERATIONAL DIAGNOSTICS ON CZECHOSLOVAK NUCLEAR POWER STATIONS

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Abstract - This paper describes the systems of operational diagnostics used on Czechoslovak nuclear power stations. The first generation systems have already been installed on nuclear power station Dukovany. Functional diagram of a system installed on the fourth unit of this nuclear power station is shown. The paper presents also a system designed for the nuclear power station Mochovce, and describes its individual subsystems. Given also are the main technical features of these subsystems. In conclusion there is presented one diagnostic situation which has occurred in the operation of the installed systems, and the analysis of measured signals is outlined.

1. INTRODUCTION

With the ever-increasing number of nuclear units in the world, the enhancing of their operational economy as well as reliability and safety is devoted more and more attention. The nuclear power stations are equipped progressively with ever-more sophisticated devices capable of providing a continuous information about the technical condition of the equipment under operation and about the technological process involved. The information about the technological process is handed over to operators with the aim of optimizing the operation, while that about the technical condition of the equipment is useful for the maintenance workers because it enables to prevent failures and accidents. In addition, this information may be of advantage in planning and preparing repairs of the equipment.

The information systems for monitoring the course of the technological process on Czechoslovak nuclear power stations are supplied in adequate quality and provide the operators with all information they need.

The systems for monitoring the technical condition of the technological equipment (i.e. the NSSS) have not been incorporated in original design. Therefore, they have been added as retrofits onto Czechoslovak nuclear power stations.

2. INSTALLING THE SYSTEMS OF OPERATIONAL DIAGNOSTICS ON CZECHOSLOVAK NUCLEAR POWER STATIONS

This activity was initiated in 1983, the first deliveries being directed to the first unit of the nuclear power station Dukovany (JEDU). Then followed supplies to further units of Dukovany nuclear power station, and to units of the V-2 nuclear power station in Jaslovske Bohunice put one after the other into operation. At the present time the equipment is being delivered to the fourth unit of Dukovany nuclear power station.

Fig. 1. presents the functional diagram of the equipment delivered to the fourth unit in Dukovany. The equipment consists of the following three main subsystems:

- a subsystem (SV) intended for monitoring vibration of primary circuit components, the subsystem being based on the analysis of signals delivered by induction-type sensors of displacements, and sensors of pressure pulsations.
Four absolute sensors of displacements are accommodated on reactor vessel flange, while each steam generator has two relative sensors of displacements. Pressure pulsation sensors are installed in the cold legs of both circuits.

- a subsystem (3C) for monitoring loose parts in the primary circuit, based on signals obtained from low-frequency accelerometers in the frequency band 1-10 kHz.
- Four such sensors are accommodated in the nozelled section of the reactor, one accelerometer is installed on each circulating pump and steam generator, seven further sensors being installed on control rod drives.
- a subsystem (3Z) for gathering data underlying residual life, the system being intended for steam generators and the pressurizer. This equipment processes temperature signals from thermocouples, vibration signals from accelerometers, as well as signals of some selected working parameters.

The analysis of signals from individual measuring subsystems has been enhanced by installing on each unit a spectral analyser with a recorder, and a measuring tape-recorder for storing the signals. This is assisted by a central minicomputer for coordinating individual subsystems and simultaneous storing of measured and analysed data.

3. MOCHOVCE NUCLEAR POWER STATION

Taking account of all aspects involved and using advantage of the experience gathered during previous installations, there has been proceeded with preparing the installation of the operational diagnostic system for the nuclear power station in Mochovce (fig. 2).

The following subsystems have been stipulated as the basic ones: the subsystem for monitoring vibrations, the subsystem monitoring loose parts, and the subsystem monitoring coolant leakage from the primary circuit. To these there have been added a subsystem monitoring acoustic signals in free space, with this subsystem processing signals delivered by measuring microphones, and the above-mentioned subsystem of monitoring residual life.

The individual subsystems contain also microprocessor units with self-contained peripheral devices, thereby enabling a simultaneous storing of measured data and results.

The following principles underlie the fundamental philosophy of the system:
- the equipment is decentralized so that individual subsystems operate independently.
- all subsystems must detect, localize, and classify the phenomenon under investigation.
- in taking time signals, there must be available also the pre-history of the occurring phenomenon.
- the equipment must enable communication with central information system of the nuclear power station.
- the system has the on-line operating mode, nevertheless, all interventions into the control and safety of the nuclear unit must perform the operator.

4. EXAMPLE OF AN IDENTIFIED PHENOMENON

Toward the end of 1986, the main circulating pump (MCP) No. 5 of the first unit in Dukovany nuclear power station displayed irregular step changes of the effective value (RMS) by some 5 dB. Time behaviour is shown in fig. 3. After ensuring that the cause was not associated with either the sensor or the transfer line, there has been performed a frequency analysis with a standard analysis of the spectrum. After performing a minor repair in December 1986 the phenomenon disappeared and, in addition, the effective value with a long time constant increased markedly, as shown in fig. 4. In the course of maintenance, we have compared the technical condition of No. 5 main circulating pump with obtained diagnostic signals. The rubber rings of glands in No. 5 main circulating pump have been exchanged. Because a similar phenomenon has occurred also in No. 1 main circulating pump, we have carried out a more detailed analysis of the spectrum with the aim of diagnostic the type of failure in No. 1 main circulating pump of the second unit, and to formulate recommendations for maintenance workers.
4.1. Arrangement of diagnostical sensors

The sensors installed on the main circulating pump are incorporated into the subsystem of monitoring loose or partly loose parts, and the signals obtained are processed in pertinent indication and processing unit. The accelerometer which serves as the sensor is situated on the main circulating pump casing flange near the bearing (fig. 5), with its axis of sensitivity being oriented vertically. The signal is originated as a consequence of the piezoelectric effect. The frequency band of the transfer line ranges between 500 Hz and 10 kHz.

4.2. Analysis of measured diagnostical signals (0-10 kHz)

The information content of the signal has been obtained by means of measurement carried out on No. 4 main circulating pump of the first unit during its shutdown. Other pumps have been operating in their nominal mode, as well as the autonomous circuits of the shut-down main circulating pump and the gland water circuit. The medium, i.e. the reactor cooling water, has been circulated in the circuit in opposite direction. The Hewlett-Packard analyser has taken spectrum from sensors 4Bl (the fourth main circulating pump) and 5Bl (the fifth main circulating pump) as shown for the first unit in fig. 6 (to make a better survey, fig. 8 contains the spectra of all main circulating pumps, as obtained during this measurement). The results obtained may be classified in the following manner:

a) Region I (0 - 1 kHz) involves typical peaks corresponding to rotating machines. Of special importance here is the rotational frequency of 24 Hz, blade frequency of 120 Hz, and their multiplies. The decrease of these signals represents for No. 4 main circulating pump up to 15 dB.

b) Region 2 (1 kHz - 1.5 kHz) is typical by its absolute independency on the operation of the main circulating pump. In other words, in this portion of the spectrum the whole output is represented by the operation of the glanding system. In our analysis, this fact may be taken advantage of in the form of being used as a selective indicator of vibration on main circulating pump glands.

c) Region III (1.5 kHz - 2.7 kHz) may be characterized in the first place by an increase of the shut-down pump signal level. Under operating conditions this region seems to be remarkably balanced, not only in the case of an individual spectrum, but also in the spectra of other, comparable devices. This is probably so due to the influence of turbulent noises and the clearance set by pump shaft rotation.

d) Regions IV and VI (2.7 - 5.8 kHz; 7.8 - 10 kHz) exhibit a broad spectral scattering. As a consequence, they have been left out of consideration.

e) Region V may be regarded as a further potential region to be observed.

On the basis of the above-mentioned fundamental properties of the spectrum we have focussed our attention to a more detailed analysis of the region 0 - 5 kHz. Fig. 7 presents spectra of No. 5 main circulating pump of the first unit, before and after outage. Performed repairs resulted in a certain shift of the band (by 2 kHz). For comparison purposes, fig. 9 presents time records of No. 5/I and No. 1/II main circulating pumps, whereas figs. 10 and 11 contain accurate spectra as got from No. 1/II and No. 5/II main circulating pumps (figs. 6, 7, 8, 12, 13 present smoothed spectra which are more convenient for comparing spectra envelopes).

4.3. Analysis of measured spectra in region II (1 - 1.5 kHz)

Signals from the main circulating pumps No. 1-6 in the second unit have been processed using the BaK analyser. In the selected sector of the spectrum we have always measured the output in the frequency band under consideration \( \frac{A}{V^2} \), in relation to a reference value of 1/V². The highest value has been measured on No. 1 main circulating pump (19 dB/const., fig. 10), i.e. just on the main circulating pump under investigation. It is the signal of this pump which causes an irregular and step change of the RMS.

\*Figs. 6 and 7: cross hatching - signal decrease after shutdown, parallel hatching - signal increase after shutdown
Another extreme has occurred on No. 6 main circulating pump (fig. 11), with TOTAL equaling to 27.7 dB/const. In the case of this signal, there occur in region II two peaks which represent the 52nd and 56th harmonical component of rotational frequency (24 Hz). The frequencies of the peaks are 1248 Hz and 1344 Hz, and their magnitude of 99.7 dB equals approximately to the extreme difference as measured in TOTAL. The occurrence in this region of rotational harmonical frequencies suggests that there exists an input of energy (f) into the glanding system, exciting vibration of glanding system components with a natural resonance nearing the corresponding harmonical components of rotational frequency.

On the basis of the approach mentioned above we have then compared in region II the spectra of No. 5/1 main circulating pump before shutdown with that of No. 1/II main circulating pump, and also No. 5/1 pump after repair with No. 1/II pump (figs. 12 and 13). The comparison suggests clearly that the type of failure which has occurred on No. 1/II main circulating pump resembles that of No. 5/1 pump. A similar conclusion has been arrived at through using the method of summation and averaging the spectra of the main circulating pumps No. 1-6 for the first unit, and the individual spectrum of No. 1/II pump (figs. 15 and 14). In the first case TOTAL amounts to -26 dB/const., whereas for No. 1/II main circulating pump TOTAL is equal to -19.5 dB/const., so that their difference of 6.5 dB exceeds the scatter of the averaged spectrum.

5. CONCLUSIONS

The experience gathered in the course of installing and operating the systems of operational diagnostics on Czechoslovak nuclear power stations justifies the approach adopted. Concerning the nuclear power station in Mochovce, the number of sensors will only be increased (e.g. in the system for monitoring vibration there will be used measuring channels for monitoring neutron flux noise, the number of relative sensors on the piping will be increased, etc), but the fundamental philosophy of the system will remain untouched.

The example mentioned hereinabove indicates that the information content of measured signals exceeds the region originally designed. A failure originating on the operating pump has been detected in our case by a sensor intended for monitoring free parts.

It has been succeeded in proving a relation between instantaneous technical condition of glands with the output and the character of spectrum in the frequency band 1 - 1.5 kHz. It has been derived from the comparison that the better is the technical condition of glands, the higher is the effect of the "rotational (dilatation) lack of sealing" of the glands. Consequentially, the lowest value of spectrum output, -27.7 dB/const., indicates the best quality of glands. Random step changes of the RMS level are attributable to the clearance of the gland which enables vibration between two stable positions and, consequently, even changes in the flowing characteristics of the gland water. In the case of No. 6/II (fig. 11) the sharp peaks in the second region may be considered for a symptom of an impaired quality of the gland. But this conclusion cannot be supported by sufficient volume of data from early period of operation.

As a result of this analysis it has been recommended to carry out, during the shutdown of the second unit, a check of the glanding system with the aim of ensuring a reliable operation of the main circulating pump in further period.
Fig. 1.
Fig. 4. Time course of signal after repair
Fig. 5. Main circulating pump
Fig. 12. Comparison of spectra envelopes

Fig. 13. Comparison of spectra envelopes
RESULTS OF NEUTRON NOISE MEASUREMENTS ON THE SUPER-PHENIX 1200 MWe LMFBR FROM START-UP TO NOMINAL POWER

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ABSTRACT — The first neutron noise measurements on SPX are presented in this paper. They were carried out throughout the year 1986 for power levels of 5 % to 80 % of nominal power. For these measurements one in-vessel detector and two under vessel detectors were used. Some frequencies around 1 Hz were always present these evolving with the operating parameters. The rms values of these resonances never exceeded half a pcm and were within the range 0.3-0.5 pcm (4 d). The interdependence of the operating parameters made it difficult to find out which one had a greater influence on the frequency evolution of the neutron noise signals. However a good candidate seemed to be the primary pump speed. The observed resonances being then originated in the fuel assemblies or control rods vibrations due to hydraulic excitation. More measurements in various operating conditions are needed to verify such a hypothesis, but the fact is that the flux signal fluctuations of SPX are low which was a point to verify at the starting period of the reactor.

1. INTRODUCTION

The only high power breeder reactor to have been under construction and run in the world, SUPER-PHENIX (SPX) started the chain reaction on September 7th 1985, was connected to the grid on January 14th 1986 and reached nominal power (3041 MWh and 1205 MWe) by December 9th 1986.

As a prototype of commercial size, this reactor has undergone a wide range of very detailed tests which can be divided into 3 stages /1/.

Stage 1: Separate tests for each elementary system and loading (February 1982 through May 1982) of the 1894 assemblies of the core, among them 386 dummy fuel assemblies which were required for the isothermal tests which were carried out at a later date.

Stage 2: Tests for the boiler as a whole before loading, and for the ex-boiler systems, they were:
- filling of secondary loops, fuel handling machine and reactor block with sodium,
- isothermal tests,
- first temperature rising test,
- second temperature rising test,
- ex-boiler test.

The first temperature rising test allowed the clarification of a vibrational problem with the internal vessels to be linked to the height of the sodium fall in the spillway (fig. 1). This problem has been identified and then overcome by increasing the cooling flow of the main vessel and to some extend changing the procedure to be followed in the (Q, T, H) diagram when starting the plant (see fig. 2). The second temperature rising test proved the validity of the solutions choosen in regard to the vibratory problem (problem stopped after 300°C).
Stage 3: Loading, neutronic tests and power rising. The loading for the starting of the fission chain reaction (325 fuel assemblies) was continued in order to obtain the core ready for the power rising (356 fuel assemblies). From the first quarter of 1986 through to the end of the year numerous tests were undertaken together with the power rising. At this stage and when the power was stable measurements were made of neutron detectors signals fluctuations.

After having reached the 100 % FN level some supplementary tests were still to be carried out. In March 1987 the problem of a sodium leak in the fuel handling machine was found and the reactor was stopped in May 1987 for the maintenance of this material.

1987 didn't permit the continuation of the neutron noise measurements on SPX and the results which are to be presented here have to be considered as a first investigation in the neutron noise signatures of this reactor. It will be seen that the interdependence which holds between several main operating parameters(*) which have an effect on neutron noise makes more measurements necessary in order to better interpret the observed signatures.

2. SENSORS

2.1. Neutron detectors

Situation

During the physical tests of the initial period and until 100 % FN, in-vessel detectors were available attached to the "BOUPHY" telescopic plug [2]. It came then as a plus in regard to the two under vessel detectors which were only to be used when the reactor was in its operating stage.

Our neutron noise measurements were made from the signals of these 3 detectors or occasionally from a reduced set of them.

- The in-vessel detectors (IVD)

The device is equipped with 3 U5 fission chambers (positions 1 to 3 in the figure /3a/). The signal of only one detector was available at any one time for neutron noise measurements the other two being connected to the safety control rod system. The three detectors were attached in the telescopic plug inserted in the central assembly /3b/, their fissile length being at a distance of 440 mm apart. They were eventually axially moved in the core. At the initial starting of the chain reaction detector n° 2 was at the very middle of the core. For a power higher than 30 % FN it was at a position of 1955 mm above the median plane. Measurements reported here were carried out with the detector in position 2 which axial location are given in table 1 hereafter.

- The under vessel detectors (UVD)

There are boron chambers located at a considerable distance from the bottom of the core and under the vessel as shown in figure /4a/.

Figure /4b/ is a detailed diagram of the under vessel detector devices. There are three of them each 120° apart. Only two have a chamber available for neutron noise. In this paper they are referred to as under vessel detector n° 1 and under vessel detector n° 2 (UVD1 and UVD2). Figure /4c/ shows the radial position of the three neutron guides in the core.

2.2. Other sensors

Also measured was the signal of some accelerometers placed on the 4 primary pumps and on the control rod drive mechanism. Figures /5a/ through /5c/ are given as example of the PSD and coherence function that were obtained.

(*) Like the flow rate Q, the primary pump speed FPS, the power P, the insertion of the control rod Hr, the inlet temperature Ti, and the delta-T in the core ΔT/.
3. MEASUREMENTS

The following table gives the conditions for the measurements made.

<table>
<thead>
<tr>
<th>Test n°</th>
<th>Neutron sensor</th>
<th>Measurement length</th>
<th>Power (% PN)</th>
<th>PPS (% NS)</th>
<th>ΔT core (% ΔTN)</th>
<th>Te (°C)</th>
<th>Height of the curtain (mm)</th>
<th>Date (1986)</th>
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</thead>
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<tr>
<td>1</td>
<td>21 1845</td>
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<td>16,25</td>
<td>37,4</td>
<td>47,3</td>
<td>358,6</td>
<td>632</td>
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<td>58'</td>
<td>4,7</td>
<td>33,2</td>
<td>14,8</td>
<td>328,0</td>
<td>598</td>
<td>13-02</td>
</tr>
<tr>
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<td>54'</td>
<td>5,9</td>
<td>35,8</td>
<td>17,9</td>
<td>331,2</td>
<td>602</td>
<td>13-02</td>
</tr>
<tr>
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<td>37,6</td>
<td>39,2</td>
<td>366,1</td>
<td>630</td>
<td>20-02</td>
</tr>
<tr>
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<td>642</td>
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<tr>
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<td>26,7</td>
<td>46,9</td>
<td>62,5</td>
<td>383,0</td>
<td>655</td>
<td>19-03</td>
</tr>
<tr>
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<tr>
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</tr>
<tr>
<td>9</td>
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<td>1995</td>
<td>280'</td>
<td>51,1</td>
<td>70,8</td>
<td>78,3</td>
<td>390,7</td>
<td>30-06</td>
</tr>
<tr>
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<td>1995</td>
<td>270'</td>
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<td>80,8</td>
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<td>1995</td>
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<td>90,7</td>
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<td>80,1</td>
<td>94,0</td>
<td>87,4</td>
<td>387,6</td>
<td>750</td>
</tr>
</tbody>
</table>

(*) The given position refers to the distance (in mm) from the centre of the sensitive length to the centre of the core. In May 86 the in-vessel chambers were up 150 mm above the previous position.

ΔT<sub>n</sub> = 150°C

PN = 3000 MWth

PPS : primary pump speed
NS : nominal speed = 452 revolution per minute.

In the operating diagram (P, Te) and (P, PPS) of fig. 6 the measurements are figured as X. There are also some points for some neutron noise results given by the authors of ref. /3/. The significance of one point is that the power was 50%, while PPS was almost at its nominal value and will be compared with our point P = 50 % PN, and PPS = 71 % NS.
4. ANALYSIS

4.1. Range 0-20 Hz

4.1.1. Under vessel detectors (UVD)

As these detectors were at a distance from the core, the detection noise level was high$^{(1)}$:

\[ 2/\tau = 8 \times 10^{-9} \text{ at } 80 \text{ } \% \text{ PN compared to } 2/\tau = 2.10^{-7} \text{ on PWR ex-core detectors signals at } P_n. \]

Very few resonances are powerful enough to be in the range of this value and hence appear on NPDS. Resonances can mainly be seen at around 1 Hz as is shown in figure 7a.

4.1.2. In-vessel detectors (IVD)

More complex signatures appear and apart from the peaks around 1 Hz it can be seen a wide resonance which evolves from 5-6 Hz with the power under 14 % Pn to 8 Hz when the power is within the range 50 % to 80 % Pn (fig. /7b/). However this phenomenon was also sensitive to the position of the sensor in its device as it is seen on fig. /7c/.

4.2. Range 0-5 Hz

Within this range are the few resonances observed on the signal of the three neutron detectors. They seem not to be linked to the detection conditions. First, from the coherence functions peaks can be seen which are common to the signals and then correspond to phenomenon to be identified but already known as to be acting on the neutron flux itself.

4.2.1. Coherence functions between neutron detectors

It can be seen in fig. /8/ and /9/ an evolution versus power and PPS of the coherence function between the 2 UVD and the couple UVD n° 2/IVD n° 2 (as an example of the general couple UVD/IVD n° 2). It is observed that a tendency of frequency sliding occurs towards higher frequencies when either power or PPS is increasing. The range for which the coherence function value is higher than 0.5 is then moving within the range (0.7-0.9 Hz) to (1.2-1.5 Hz). In figures /10/ and /11/ the peak value of the coherence functions versus PPS are represented. A tendency toward a linear behaviour between the two parameters is likely but more measurements are needed to verify this.

Comparing figures /8/ and /9/ it can furthermore be seen that some peaks in figure /8/ are not in figure /9/ showing then that the fluctuations received by the UVD and IVD detectors are only a part of that which is measured individually by each.

4.2.2. NPDS

The NPDS of UVD n° 2 in the range 0-5 Hz are superimposed for various power levels in fig. /12/. The same has been done for IVD n° 2 in fig. /13 a-b/. Resonance peaks that can be pointed out are more numerous than on coherence functions, illustrating then the previous remark according to which the detector signal seems to undergo fluctuations which are not always in their whole propagated in the total under flux area.

An evolution tendency of peak frequency values with PPS can also be seen (see fig. /14a/ and /14b/).

4.2.3. Phases laws

When meaningful (i.e for the ranges of coherence function $> 0.5$), the phases values between all couples of detectors signals is 0, as can be seen in figure /15 a-b/.

4.2.4. Coherence functions between neutron sensors accelerometers and thermocouples signals

Such functions have been calculated for some pumps and control rod accelerometers. In general no coherence can be pointed out (see fig. /5 a-b-c/). According to /3/ none coherence is obtained between thermocouple signal fluctuations and neutron noise.

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$^{(1)} \tau = $ detection rate.
4.3. Outstanding observations

Hereunder are the most important points that can be drawn in order to sum up our observations:

- peaks around 1 Hz at which the damping ratio is ≥ 8 %,
- there are seldom harmonics,
- sensitivity versus the PPS of the frequencies near 1 Hz,
- peaks were not always present on the signal of both types of detectors (IVD and UV) which in fact corresponds to two locations (one far below the core, the second a short distance above it),
- wide peak around 6-8 Hz on IVD only,
- specific measurements in cold conditions showed that first eigenmode was around 1.7 to 3 Hz for the fuel and around 1.6 Hz for the control rods.

5. INTERPRETATION

The set of measurements which were carried out in 1986 have to be considered as a verification that no powerful reactivity fluctuations were present and as a determination of SPX neutron noise signatures.

The first point has been reached since no powerful values for the NPSD resonances have been found. The corresponding rms values for IVD are generally in the range of 50 % to 60 % of the total rms values and without considering the detection noise (on the 0-20 Hz range). At 80 % P_N the detection noise is about 5 % of the total noise (0-20 Hz) while it is about 14 % when P = 14 % P_N.

For UV the corresponding values are in the range of 40 to 70 % of the total rms value (0-20 Hz) which is composed at P = 80 % P_N by 80 % of detection noise while the ratio is 91.5 % for P = 14 % P_N.

Concerning the measured rms values in pcm with \( B_{\text{eff}} = 380 \) pcm, and the hypothesis of being in the plateau region, the measured reactivity fluctuation (at 40) has never been higher than half a pcm and usually layed in the range 0.3-0.5 pcm.

Concerning the second point, the measurements cannot yet allow a precise interpretation of the observed frequencies which are marked in figures /10/, /11/ and /14a-b/. Actually, the operating rules of an LMFBR are such that several parameters influencing the neutron noise evolve together. This is the case for the power, the axial position of control rods curtain, and the primary pump speed.

However, the comparison of the neutron signals in the rare cases which differed by only one parameter (P = 50 %, PPS = 71 % and P = 50 %, PPS = 94 %) indicate that PPS seems to be the most important parameter for frequency evolution.

Considering this hypothesis frequency values were connected suggesting then an interpretation in term of linear frequency evolution versus the PPS (see fig. /16/).

The concerned frequency region (1 to 1.5 Hz) mainly suggests reactivity effects due to fuel assemblies or control rods which first eigen frequencies in cold conditions were found to be around 1.6, 1.7 Hz. Such an interpretation which can't at present be quite definite is also in agreement with the survey made in /4/ relative to the neutron noise in all LMFBR operating in different countries (excluding USSR), whatever size of reactor was considered. Comparing the different experiments it has been reported in /4/ that neutron noise in operating LMFBR has been, up to now, mostly if not for sure exclusively, due to control rod or fuel assemblies motion.

Besides these frequencies, the range 5-8 Hz of the very damped resonance observed on IVD can't yet be accurately interpreted since it has been showed to be sensitive either to the power (or other operational parameters) or to the position of the neutron sensor in the BOUPHY-plug.
Concluding remark

Throughout the year 1986 neutron noise measurements were made from IVD and UVD. This work was aimed to assess the reactivity fluctuations and verify that they can't be originated by some abnormal behaviour of SFX under starting. It was found that the level due to the phenomena other than detection noise was less than 0.5 pcm which is obviously low. An other aim was to determine SFX neutron noise behaviour and to find a precise interpretation of the measured signatures. A set of frequencies around 1 Hz were found as the main characteristic with a likely linear behaviour of the peak values versus PPS. But due to the strong dependency in the evolution of the main operating parameters that can't be assessed before further measurements in various operating conditions.

REFERENCES

/1/ RGN Avril 87 n° 2 Mars-Avril.
/2/ J.C. NERVY and al "Experimental devices used for start-up operations of the SUPER-PHENIX core" International conference on fast breeder systems. Richland USA September 13-17 1987.
252 pins Ø8.5
wire O.D 1.2.
capsule

Fig. 3-b SPECIAL FUEL SUB-ASSEMBLY
CROSS SECTION
control rods

in-vessel detectors
(IVD)

under vessel
electrons detectors
(UVD)

Neutrons guide

Figure 4-a :
N.G.

N.G.

N.G.

N.G. neutrons guide

Figure 4-b

+1485
+1215
+995
+785
polystyrene

helium detector

ionisation chamber
used for neutron noise
fission chambers

neutrons guide axe

Figure 5-a : 14/3/66 PPE 7 26.6% 383.1 476.1 212 revolution per min

Figure 5-b : 14/3/66 RBC 11 26.6% 383.1 476.1 212 revolution per min

Figure 5-c : COHERANCE FUNCTION BETWEEN RBC 11
AND NEUTRON SIGNAL NO5

2.65
Figure 6

Figure 7-a: SUPERPOSITION OF UVD N°2 NPSD AT VARIOUS POWER LEVEL

Figure 7-b: IVD n°2 AT DIFFERENT OPERATING CONDITIONS AND AT THE SAME AXIAL POSITION (1983MM ABOVE THE AXIAL MEDIANE OF THE CORE)

Figure 8: COHERENCE FUNCTION BETWEEN THE TWO UNDER-VEssel SIGNALS

1: $P = 80\% P_N$, PPS = 425 r/mn
2: $P = 70\% P_N$, PPS = 410 r/mn
3: $P = 60\% P_N$, PPS = 365 r/mn
4: $P = 50\% P_N$, PPS = 320 r/mn
5: $P = 31\% P_N$, PPS = 231 r/mn
Neutron noise measurements on SPX

Figure 9: COHERENCE FUNCTION BETWEEN ONE UNDER-VESSEL SIGNAL AND THE IN-VESSEL ONE.

1: P = 80% P_N, PPS = 425 r/mn
2: P = 70% P_N, PPS = 410 r/mn
3: P = 50% P_N, PPS = 320 r/mn
4: P = 14% P_N, PPS = 198 r/mn

Figure 10: FREQUENCIES OF THE MAXIMA FOR THE COHERENCE FUNCTION OF THE TWO UNDER-VESSEL SIGNALS, VERSUS THE PRIMARY PUMPS SPEED (PPS)

Figure 12: UVN N°2 NPSD AT VARIOUS OPERATING CONDITIONS (RANGE 0-5 Hz)

Figure 13-a: UVN N°2 AT VARIOUS OPERATING CONDITIONS AT LEVEL 1845 mm 0-5 Hz RANGE

Figure 13-b: UVN N°2 AT VARIOUS OPERATING CONDITIONS AND AT LEVEL 1995 mm 0-5 Hz RANGE
VIBRATION

Session chairman: D. Wach (F.R.G.)
SUMMARY OF THE SESSION

Epstein presented the results of theoretical and experimental investigations aimed at the determination of the vibrational behaviour of reactor vessel internal structures of French PWRs. Of particular interest were the investigations of fractures of one or more flexures (supports) connecting the cylindrical thermal shield with the core barrel as well as investigations of effects of changes in the hold-down springs at the reactor flange. The computation was based on a hydroelastic model using subroutines of the CASTEM program system, the experimental simulations were performed with the SAFRAN loop, a 1:8 scaled representative hydroelastic mock-up. Computational and experimental results were in good agreement. Using these results shell mode vibrations of the thermal shield as measured in the ex-vessel neutron noise can be better monitored and evaluated.

Wehling described a new microprocessor-based vibration monitoring system (SUS-86) as installed in the newest KNU "Convoy" plants in FR Germany. Frequency and amplitude at resonance frequencies are monitored with respect to deviations from reference spectra. A quotient spectrum from the current and the reference vibration and noise signals is calculated and compared with thresholds in order to determine abnormal deviations. Up to 48 dynamic signals are evaluated simultaneously. Further developments are directed towards expert systems providing the user with direct access to expert knowledge.

Lievens discussed a new method to measure core barrel motions of the WWER-440-DWRs in GDR. Excise neutron noise signals are correlated with the envelope of at least one acoustic signals measured at the outer surface of the reactor vessel. The sound if generated by the leakage cross flow from the coolant inlet directly to the outlet. Significant coherence between the neutron noise and the acoustic signals is found as long as the direction of the core barrel vibration is not orthogonal to the acoustic sensor position.

Trenty presented results of acoustic and incore neutron noise investigations at the French 1300 and 900 MW reactors performed in order to characterize the phenomena of thimble vibrations and shocks of the instrumentation against their guides producing wear and even leakage. Studies of the statistical distributions of burst amplitudes, impulse rates of shock etc. were performed. Clear correlations were found between shocks and neutron noise fluctuations enabling the estimation of the thimble modal shape in the instrumentation tube of the assemblies. In a new plant all incore guide tubes have been equipped with accelerometer and an online monitoring system which transmits the main shock parameters to the central analysis center of EDF. Akerheim reported on incore neutron noise measurements in a Swedish BWR. The analysis gave indications that instrument tube vibrations occurred in several cases. A phenomenological model for the incore spectra has been developed by including a global resonence-like term and its first harmonic into the noise sources representing the detector vibration itself within a linear and a curved neutron flux distribution. Comparisons of model results and measurements analyses showed good qualitative agreement.

Quinn presented two papers dealing with thermal shield problems in several CE nuclear reactors. On behalf of his colleague Dr. Lubin he presented results of reanalyses of the internals vibration and loose parts monitoring systems data over the operating lifetime of the St. Lucie Unit 1 plant. During the refueling after the 5th fuel cycle the thermal shield was found to be damaged. Dynamic analysis of the core barrel - thermal shield system was utilized in the interpretation of the reanalyses of the excise neutron noise and accelerometer signals. In his second paper Quinn described an equipment and a surveillance program (developed together with the utility) which monitors the performance of the thermal shield support structure at the Fort Calhoun nuclear station. Neutron noise analyses of ex-vessel detectors were performed near zero reactor power to determine the first shell mode frequencies of the core barrel-thermal shield system. Since the identified frequency was stable in later measurements, the new support structure was found to be effective.
MALFUNCTION TESTS AND VIBRATION ANALYSIS OF PWR INTERNAL STRUCTURES

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Abstract - To diagnose changes liable to occur in the vibration behavior of internals, it is important to understand the influence of changes in the mechanical properties of elements on the output signals obtained from neutron chambers placed out of core and accelerometers fixed to the reactor vessel. To do this, the effects of changes liable to occur in the hold-down springs and the flexures were simulated on the SAFRAN loop, using a representative hydroelastic mock-up (geometric scales 1/8). The results obtained experimentally on SAFRAN for different characteristics of the hold-down spring, which lies between the upper part of the core barrel and the vessel head, have been published.

In this paper, we propose to present the results of the investigation of the fracture of one or more flexures which connect the cylindrical thermal shield to the core barrel. This work is in two parts:

a) Computation based on a hydroelastic model using the sub-structure computer program TRISTANA of the CASTEM system.

b) Tests simulating flexure fracture:
   1 - in air, for an understanding of the mechanisms involved;
   2 - on the SAFRAN loop with a representative flow in order to estimate the strains liable to exist on the vibration signatures recorded on displacement transducers and accelerometers.

Good agreement was observed between the computation results with the theoretical model employed and those obtained experimentally.

In this presentation, we shall provide details showing that the detection of the occurrence of this process is very delicate. However, on the nuclear power spectral densities obtained on a power plant, it seems possible to detect a light frequency shift of the shell mode on the thermal shield in certain operating situations. This study, which required substantial experimental and computation resources, was conducted as a joint project between Commissariat à l’Energie Atomique, l’Electricité de France and Framatome.

COMPORTEMENT VIBRATOIRE DES STRUCTURES D’INTERNES DEGRADEES DE R.E.P.

1. INTRODUCTION
Les vibrations des internes provoquent des modifications des fluctuations neutroniques recueillies sur les chambres d’ionisation placées autour de la cuve. Une interprétation de celles-ci nécessite une bonne connaissance des phénomènes mécaniques à l’origine de ces perturbations sur les DSP (Densité Spectrale de Puissance) de bruit neutronique.
Depuis 1972, une série d'études concernant le cas des Réacteurs à Eau sous Pression (REP) de type à 3 boucles de refroidissement de puissance électrique 900 MWe, a été réalisée ; celle-ci peut se diviser en trois grands thèmes :

1) Étude théorique, expérimentale, sur maquette représentative à échelle réduite hydroélastique sur la boucle SAFRAN et mesures pendant les essais à chaud sur les centrales Fessenheim 1, Tihange et Bugey 5 (écran thermique cylindrique) et Tricastin (écran thermique à secteurs) pour déterminer les caractéristiques du comportement vibratoire, normales des internes.

2) Étude théorique et uniquement de simulation sur maquette de la dégradation des anneaux de calage.

3) Une dernière étude comportant des calculs de fréquences et des déformées modales d'enveloppe et d'écran thermique associés à des essais en air et sur la boucle SAFRAN sur une maquette simulant quelques cas de dégradations de supports flexibles.

Nous nous proposons de rappeler succinctement les conclusions se déduisant de la simulation de la perte progressive des propriétés mécaniques de l'anneau de calage et de celles concernant les dégradations des supports flexibles.

Nous examinerons aussi les éléments pouvant servir au diagnostic de l'apparition de tels phénomènes. Nous essaierons d'extrapoler les conclusions au cas des DSP de bruit relevés sur une centrale de cette famille de REP*.

2. ETUDE DU COMPORTEMENT VIBRATOIRE DES INTERNES

Le comportement vibratoire des internes de REP est réglé par des phénomènes complexes dont la compréhension a nécessité des analyses détaillées. Une étude a été entreprise par le DMT** en collaboration avec la Société Framatome et l'EDF***.

2.1. Conditions normales et anneau de calage dégradés

2.1.1. Généralités. La surveillance en service des REP a montré que la signature vibratoire pouvait être sujette à des variations au cours du temps, en particulier en ce qui concerne les mouvements de balancement des internes. L'expérience acquise sur modèles réduits (essais SAFRAN) et l'analyse des fonctionnements sur site, à l'étranger notamment, montrent que les variations les plus notables sont liées à des modifications de conditions aux limites de l'enveloppe de coeur, à savoir :

- suppression des jeux (blocage des guides radiaux au niveau inférieur de celle-ci ;
- défaut de serrage de l'anneau de calage en partie supérieure.

La première cause en principe ne présente pas de caractère gênant du point de vue vibratoire car elle entraîne une raideur accrue donc un niveau vibratoire plus faible des internes.

La seconde est liée à une dégradation de l'anneau de calage qui peut être plus progressive. Celle-ci peut devenir dangereuse du point de vue vibratoire car elle se traduit par une plus grande souplesse des internes.

2.1.2. Étude théorique [2] [3]. La dégradation des anneaux de calage se traduit par l'introduction d'une rotule à la place d'un encastrement entre bride et support de cuve et par une diminution de la raideur axiale aux mêmes endroits.

Ces évolutions de caractéristiques sont difficiles à estimer ; toutefois, les différents calculs ont montré que dans tous les cas, la fréquence du mode de balancement des internes diminue avec l'importance de la dégradation.

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* REP : Réacteur à Eau sous Pression.
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*** Electricité de France.
2.1.3. Etude expérimentale [2] [3]. Les essais ont été effectués sur la boucle SAFRAN (Fig. 1 et 2 de la référence [1]). La figure 1 donne le détail de la simulation des anneaux de calage. La figure 2 montre les variations de la DSP obtenue à partir d'un capteur de déplacement placé entre l'enveloppe de cœur et la cuve pour quatre anneaux différents. Nous constatons que la fréquence correspondant au mode de balancement des internes diminue avec la dégradation comme le laissait prévoir le calcul, avec une augmentation d'amplitude et de l'amortissement. Sur la figure 3 est représentée une DSP se déduisant d'un signal recueilli sur un accéléromètre placé sur le fond de la cuve. Nous remarquons que la fréquence correspondant au balancement de l'ensemble cuve-internes n'est pas affectée sensiblement par une dégradation de l'anneau de calage.
2.2. Flexibles dégradés (concernant les réacteurs à écran cylindrique - 6 réacteurs du parc EDF à Fessenheim et à Bugey).

2.2.1. Généralités. L'étude du comportement vibroïde des internes avec des liaisons flexibles endommagées a été menée de façon identique aux précédentes. Elle a consisté en un calcul utilisant un logiciel pour sous-structure "TRISTANA" développé par le CEA. Elle a été complétée par deux séries d'essais sur maquette à échelle réduite sur laquelle nous pouvons simuler la dégradation des flexibles. La première en air avec une excitation sinusoidale appropriée a servi à avoir une meilleure compréhension des phénomènes, la deuxième sur la boucle SAFRAN est destinée à estimer les déformations sur les signatures vibratoires pour plusieurs cas d'endommagement.

2.2.2. Etude théorique. De nombreuses combinaisons de ruptures de flexibles ont été calculées.

La rupture d'une ou plusieurs attaches entraîne des glissements des fréquences caractéristiques des différents modes.

Nous donnons que les résultats du mode 2 d'écran thermique qui possède une fréquence diminuant avec le nombre de flexibles rompu.

<table>
<thead>
<tr>
<th>Normal</th>
<th>autour de 11,2 Hz</th>
</tr>
</thead>
<tbody>
<tr>
<td>1 attache rompue</td>
<td>9,8 Hz</td>
</tr>
<tr>
<td>2 attaches rompues (éloignées)</td>
<td>9,4 Hz</td>
</tr>
<tr>
<td>2 attaches rompues (voisines)</td>
<td>8 Hz</td>
</tr>
<tr>
<td>3 attaches rompues (voisines)</td>
<td>7 Hz</td>
</tr>
<tr>
<td>4 attaches rompues (voisines)</td>
<td>6,6 Hz</td>
</tr>
</tbody>
</table>

2.2.3. Etude expérimentale. Le schéma de la figure 4 indique les positions respectives des liaisons flexibles entre l'enveloppe de cœur et l'écran thermique et la figure 5 une photographie d'une attache de la maquette.

a) Essais en air : Ceux-ci ont permis de constater que la suppression d'une et deux attaches se traduisait par des déformations appréciables sur les DSP obtenues à partir de signaux d'accélérométrie. En particulier pour le mode de l'écran une modification de deux fixations entraîne une diminution importante de la fréquence et un élargissement de la résonance (figure 6).

Un essai avec un flexible rompu volontairement n'a pas montré d'apparition de chocs, c'est-à-dire de signaux à plus haute fréquence.

b) Essais sur SAFRAN : Les résultats relevés sur des capteurs de déplacement placés entre écran et enveloppe et entre écran et cuve extérieure montrent des modifications très nettes des DSP en cas de disparition de deux flexibles.
Fig. 4 - Position des attaches entre écran thermique et enveloppe coeur

Fig. 5 - Flexible entre enveloppe coeur et écran thermique
A titre d’illustration, les figures 7 et 8 donnent des exemples de courbes obtenues expérimentalement. Nous avons ainsi constaté une diminution de certaines fréquences de résonance et de la déformation de celles-ci en cas d’avarie aux liaisons. Des accéléromètres placés à l’extérieur de la cuve ont confirmé que le mouvement d’ensemble de balancement des internes n’était pas affecté par la rupture d’un ou deux flexibles.

Fig. 6

Fig. 7 et 8 - Comparaison des densités spectrales "structure intègre et structure dégradée" - Excitation sous écoulement pour deux capteurs situés entre enveloppe coeur et écran thermique (deux flexibles voisins non en place)
3. ELEMENTS POUR LA SURVEILLANCE

Les études tant théoriques qu'expérimentales permettent de dégager quelques règles permettant de caractériser, sur les DSP obtenues à partir de signaux recueillis sur des chambres d'ionisation hors coeur, les phénomènes. Le tableau 1 résume ces connaissances [5] [6].

Tableau 1. Quelques dégradations d'internes

<table>
<thead>
<tr>
<th>Causes possibles</th>
<th>Bandes de fréquences</th>
<th>Variation</th>
<th>Remarques</th>
</tr>
</thead>
<tbody>
<tr>
<td></td>
<td>Etat normal</td>
<td></td>
<td></td>
</tr>
<tr>
<td>Contact au niveau des guides radiaux</td>
<td>De 7 à 11 Hz</td>
<td>Sans contact :</td>
<td>1) Phénomène pouvant être intermittent</td>
</tr>
<tr>
<td></td>
<td>(Mouvement de l'enveloppe coeur n = 1)</td>
<td>7/8 Hz.</td>
<td>2) A été constaté sur centrale.</td>
</tr>
<tr>
<td></td>
<td></td>
<td>Avec contacts :</td>
<td></td>
</tr>
<tr>
<td></td>
<td></td>
<td>10/11 Hz.</td>
<td></td>
</tr>
<tr>
<td>Dégradation des anneau de calage</td>
<td>Normal 7/8 Hz</td>
<td>Avec dégradation :</td>
<td>Le suivi du 1er mode permet une détection aisé d'une dégradation.</td>
</tr>
<tr>
<td>(Ensemble des réacteurs du parc EdF)</td>
<td></td>
<td>Diminution de la fréquence.</td>
<td></td>
</tr>
<tr>
<td></td>
<td></td>
<td>Elargissement de la résonance.</td>
<td></td>
</tr>
<tr>
<td>Dégradation des flexibles (bas d'enveloppe coeur)</td>
<td>10 à 22 Hz</td>
<td>Modification des résonances dans certaines bandes en particulier la fréquence n=2 d'écran thermique diminue.</td>
<td>La modification des DSP indique une évolution qui demande une étude particulière pour proposer un diagnostic.</td>
</tr>
<tr>
<td>Réacteurs Fessenheim et Bugey à écran cylindrique</td>
<td>Modes coques (n=2) de l'enveloppe de coeur et de l'écran thermique</td>
<td></td>
<td></td>
</tr>
</tbody>
</table>

3.1. Cas particulier des flexibles dégradés

Le problème du diagnostic est plus délicat ; une déformation de la DSP peut indiquer une rupture des flexibles. Les auteurs de [4] ont constaté expérimentalement sur des REP américains que les résonances correspondant aux modes de coque avaient tendance à diminuer. Pour vérifier si ces résonances sont représentatives des modes supérieurs de vibrations, il est indispensable de disposer des DSP relevés entre deux détecteurs placés à 90° et 180°.

3.2. Diagnostic

Pour les contacts au niveau des guides radiaux et la dégradation des anneau de calage, des règles simples peuvent être écrites et utilisées pour un diagnostique automatique. Pour le cas de flexibles dégradés, le problème est plus délicat. Dans l'état actuel de l'art, une réflexion approfondie est nécessaire pour essayer d'extraire de la masse des résultats théoriques expérimentaux et de l'extrapolation des mesures sur centrale d'un type à celle d'un autre type, des lois facilement implantables sur un système automatisé.

4. CONCLUSION

Les nombreuses études menées sur le comportement vibratoire des internes montrent qu'il est possible, à partir de l'analyse des fluctuations neutroniques recueillies sur les chambres hors coeur, de détecter aisément une anomalie ou un défaut.

Pour quelques cas usuels, il est aisé d'automatiser le diagnostic, pour le cas de dégradations de causes diverses ou celui par exemple de cassures de flexibles, un effort est à faire pour extraire des règles simples. Il sera ainsi utile d'utiliser les connaissances obtenues sur des défauts observés sur des centrales et d'affiner les méthodes d'extrapolation à d'autres de type voisin.

Sur la base de ces résultats, il est envisagé de renforcer la surveillance des réacteurs à Écran Circulaire en suivant particulièrement le Mode n = 2 de l'écran.
thermique par des techniques de traitements croisés entre capteurs en temps réel sur site.

REFERENCES


VIBRATION MONITORING OF LIGHT WATER REACTORS WITH ADVANCED METHODS AND THE NEW MICROPROCESSOR-BASED SUS-86 SYSTEM

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Abstract - Damage to mechanical structures can be detected as deviations in vibration behaviour. This is particularly the case with frequency and/or amplitude deviations at points of resonance. The new SUS-86 vibration monitoring system monitors vibration behaviour automatically and documents the results in the form of a report. Its main function is to monitor discrete components on the basis of discrete frequencies within the frequency range between 0 and 200 Hz. Overall amplitude monitoring is carried out to check whether new resonance points have appeared or old ones disappeared.

1. INTRODUCTION

Vibration monitoring is performed in different ways throughout the world on a very large variety of machines and components. From the customer's point of view the results obtained frequently do not satisfy expectations. Reasons for this may include:
- Inadequate possibilities for measurement and/or inadequate measuring equipment
- Ininsufficiently clear presentation of the results in a form comprehensible only to the expert
- Lack of unequivocal evaluation criteria

Therefore, before commencement of vibration monitoring, a check should be made whether, for example, the following conditions for successful performance of the work have been fulfilled:
1. That the VIBRATION BEHAVIOUR in the new/reference condition of the object to be monitored is known.
2. That a suitable monitoring SYSTEM is available.
3. That the METHODS used are adapted to the special problem.
4. That the monitoring CRITERIA used permit consistent and reproducible evaluation of deviations in vibration behaviour.
5. That the necessary CONCLUSIONS can be drawn if unacceptable deviations exist.

KWU's new SUS-86 vibration monitoring system is based on twenty years' experience in the field of vibration measurement and monitoring and takes into account the five conditions mentioned above. SUS-86 will be used for the first time in the KWU Convoy plants (1300 MW FWR plants: Isar 2, Emsland and Neckarerweinheim 2). The system is software-oriented and features interactive user communication.

2. MONITORING METHOD

The primary function of the SUS system is to monitor discrete components. It does this by comparing the results of current vibration measurements with those performed previously. SUS-86 employs the principle that the dynamic properties of a component are particularly pronounced under resonance conditions.
In presenting the results in the frequency domain, the points of resonance are shown as peaks which can be fully described by the parameters frequency $f$, amplitude $A$ and damping $D$.

In vibration monitoring pursuant to Nuclear Safety Standard KTA 3204, frequency is the most important parameter for a number of reasons. In the monitoring process, frequency must be given preference because it is dependent neither on the location of the monitoring sensor nor on the sensitivity of the instrumentation chain. This is not true of the vibration amplitude. For the purpose of vibration monitoring, damping can only be considered a parameter which provides supplementary but not fundamental information.

2.1 Discrete Frequency Monitoring

It must therefore be concluded from the aforementioned that since the main function of the SUS-86 system is to monitor discrete components, the dominant method employed must be the monitoring of discrete frequencies.

The SUS-86 system accepts inputs on
- COMPONENTS to be monitored, their RESONANCE FREQUENCIES and MODE SHAPES /2/,
- changes in resonance frequencies with OPERATING CONDITION /3/,
- storage of REFERENCE MEASUREMENTS for comparison,
- the system is designed such that movements of the internals of tanks and vessels can be MEASURED on the OUTSIDE of the tank or vessel /4/.

The monitoring process comprises eight steps /3/: measurement, transformation, marking, identification, normalization, comparison, evaluation, deduction. The first seven steps are carried out automatically by the SUS-86 system. Currently, the eighth step must still be performed by a vibration expert. The steps measurement through normalization are adequately described in the literature /5, 6, 7/.

With regard to "comparison", a frequency range with lower and upper frequency window limits $f_l$ and $f_u$ is first fixed within a spectrum in which the vibration behaviour of a discrete component is best represented; in this frequency range the SUS-86 system automatically determines the maximum value of the amplitude (Figure 1) and enters the associated frequency in a table (see Table 1) for documentation purposes. The actual frequency determined is then converted to the operating condition at which the reference measurements were taken.

![Figure 1: Monitoring of discrete Frequencies](image)

The difference between the normalized frequency and the comparison frequency is the frequency deviation, which is then compared with the tolerated frequency deviation. In state-of-the-art monitoring technology, the tolerated frequency deviation must be defined for each individual component. To do this, information can be obtained from expedient sources, e.g.

- from the manufacturer of the component itself, in the form of allowable frequency deviation data,
or in the form of empirical frequency deviation data measured during operation without the occurrence of damage to the component.

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</thead>
<tbody>
<tr>
<td>001</td>
<td>L4</td>
<td>T.n</td>
<td>B4D</td>
<td>308.0±7.000</td>
<td>1.3856</td>
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<td>L2</td>
<td>T.n</td>
<td>R2D</td>
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<td>2.6010</td>
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<td>1.7</td>
</tr>
<tr>
<td>001</td>
<td>L3</td>
<td>T.n</td>
<td>R3D</td>
<td>308.0±1.000</td>
<td>2.2556</td>
<td>2.6010</td>
<td>0.4949</td>
<td>1.7</td>
</tr>
<tr>
<td>001</td>
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<td>R4D</td>
<td>308.0±1.000</td>
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<td>2.6010</td>
<td>0.4949</td>
<td>1.7</td>
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<td>2.2556</td>
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<td>B4E</td>
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<tr>
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<td>B4E</td>
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<td>5.3894</td>
<td>4.6892</td>
<td>4.0071</td>
<td>1.7115</td>
<td>1.4</td>
</tr>
</tbody>
</table>

Table 1: Monitoring List (test example)

2.2 Overall Amplitude Monitoring

In addition to monitoring discrete peaks it is also advisable to monitor, for example, the appearance of any new peaks or the disappearance of any existing peaks.

The quotient of "current" amplitudes and "reference" amplitudes is also calculated by the SUS-86 system and shown in the frequency domain.

In Figure 2, changes in frequency, amplitude and damping are clearly shown as characteristic double or single peaks. The criterion of evaluation is the tolerated amplitude deviation which can be defined as a constant or variable value.
Figures 3 and 4 show two typical examples of overall amplitude monitoring spectra. Two auto power density spectra (PSD) are shown in Figure 3a, the amplitudes of which differ only slightly from each other. If the quotient PSD1/PSD2 is taken, a quotient spectrum is obtained as in Figure 4a: the curve runs within a narrow horizontal band which can be used to define the tolerated range of deviation. Two auto power density spectra are shown in Figure 3b which have untolerable amplitude deviations. On the associated quotient spectrum the shaded areas are those in which the amplitude deviations exceed tolerated limits.

The SUS-86 system automatically determines untolerated deviations and informs the user of the frequency ranges in which they occur.

![Fig. 3](image1.png)

**Fig. 3**

3a: Two Spectra PSD 1 and PSD 2 with Tolerable Amplitude Deviations
3b: Two Spectra PSD 1 and PSD 2 with Non Tolerable Amplitude Deviations

![Fig. 4](image2.png)

**Fig. 4**

4a: Quotient Spectrum PSD 1/PSD 2 with Tolerable Amplitude Deviations
4b: Quotient Spectrum PSD 1/PSD 2 with Non Tolerable Amplitude Deviations

3. SUS-86 HARDWARE

The structure of the hardware is shown in Figure 5. The transducers are described in the literature /5, 6/. Up to 48 dynamic signals can be acquired simultaneously. In addition, up to 128 quasi-static operating signals can be manually input into the SUS-86 system either in analog or digital form.

The main components of the signal processor are bit slice microprocessors and high-speed multiplier-accumulator modules for digital real-time signal processing and filtering as well as spectrum analysis at the operating rate of 25 mega operations per second.

The host computer is an Intel 310-AH42 with an 80286 processor. For data input/output and storage, the system includes an alphanumeric screen, a graphics screen, a digital tape unit, Winchester, floppy and laser printer.

As an option, the SUS-86 can be fitted with a rotary equipment monitoring module (PUM) (e.g. for detecting damage to the shafts of reactor coolant pumps).
Figure 6 shows the spatial arrangement of the electronic components in two switchgear cabinets.

![SUS-86 Hardware Structure](image)

**Fig. 6: SUS-86 Hardware Structure**

### 4. SUS-86 SOFTWARE

The operating interface of the software is structured in a clear manner and can be compared with the structure of a book (Figure 7). It consists of chapters, subchapters and individual pages. After system initialization, the user can, depending on his level of knowledge and specific objectives, read all the pages or dispense with the reading altogether and restrict his activities to starting the monitoring process and collecting the results. Chapters are dedicated to key topics, such as:

**Terms and Abbreviations:** The user is familiarized with the designations used by the system for components, mode shapes, signals and evaluation functions.

**Reactor:** This chapter lists information on the name, type, history and actual operating condition of the reactor.
Measurement: Information is stored here on measurements/evaluations which have already been performed. Other features in this section include commencement, adjustment, calibration and measurement.

Vibration Monitoring: In this chapter the user obtains a monitoring list and information on, for example, whether any unacceptable frequency deviations have occurred.

Evaluation: The possibilities of special evaluations or the standard evaluation are displayed and started.

Data: The existing/acquired data in the system can be displayed on the screen and used as required for evaluation purposes.

Information: The user receives general information regarding vibration monitoring, the SUS-86 system as well as help regarding the individual menu sheets.

Fig. 7: Software-Structure of SUS-86

5. CAPABILITIES OF SUS-86

The SUS-86 system can be adapted to diverse applications. On the one hand it can automatically perform the standard vibration monitoring pursuant to KTA 3204 as described in Section 2; on the other hand it can be used generally as an easy-to-use measuring and evaluation device for vibration measurements. As already mentioned, the system simultaneously acquires up to 48 dynamic signals and evaluates them in parallel. For 42 channels the Fourier analysis is carried out in the ranges 0 to 50 Hz and 0 to 200 Hz; six channels can be measured and evaluated in the range 0 to 10 kHz parallel to the aforementioned ranges. The dynamic signals can be represented in the time, frequency and amplitude domain; all functions necessary for vibration monitoring/measuring are available in the SUS-86 system, e.g. time signal, auto power density spectra, coherence spectra, transfer spectra, amplitude distribution, etc. If necessary, the software can readily be extended. The SUS-86 system can calculate the spectral values of up to 20 correlation functions of each measuring operation. The spectral values are formed on the basis of 100 individual spectra.

5.1 Standard Vibration Monitoring

Standard vibration monitoring including adjustment and calibration of the instrumentation chains is performed automatically. It takes about 8 minutes to perform the measurement and transformation and to store the associated data onto the hard disk. In a further four minutes up to 200 individual peaks can be identified from 120 spectra, the associated frequencies normalized and compared with
reference frequencies and any deviations evaluated. After approximately 15 minutes, a standard report is available, which comprises the following:
- standard cover sheet
- reactor operating history
- SUS-86 operating history
- operating condition
- condition of instrumentation chain
- monitoring lists
- current spectra.

The main body of information in the standard report is contained in the monitoring lists and addresses the question whether the frequency and amplitude deviations are tolerable or not. If there are deviations which are intolerable, a vibration monitoring expert should be called in to draw the necessary conclusions.

5.2 General Vibration Measurements

The SUS-86 system permits modifications of the standard vibration monitoring system (Figure 8) in the case of both signal acquisition and signal processing. Other instrumentation chains, for example, (transducer plus amplifier), can be connected to the SUS-86 system (Figure 9). A further variant is the input of analog data stored on magnetic tape (Figure 10). The evaluation process in the SUS-86 provides maximum flexibility in the shortest possible time. The results can be represented in the time, frequency and amplitude domains or be shown as trends. Up to 8 signal/status representations are possible in one diagram. Various forms of diagram can be chosen: individual, overlapping, displaced or cascade. Before preparing a diagram, information must be input in accordance with Table 2. All input data can be stored and called up again. In wake-up condition the data available in the outlined boxes automatically appear on the alphanumeric screen. Some typical evaluation examples are shown in Figures 11 to 16.
Fig. 11: PSD-Spectrum with the Results: Marking, Identification

Fig. 12: PSD- and Phase Spectrum: Arrangement with Overlapping

Fig. 13: Four PSD-Spectra: Arrangement with Overlapping

Fig. 14: Four PSD-Spectra Single Arrangement

Fig. 15: Four Phase Spectra: Arrangement with Overlapping

Fig. 16: Four Coherence Spectra: Single Arrangement
5.3 Monitoring of Rotary Parts

As indicated in Section 3 above and in Fig. 5, addition of a special module allows the SUS-86 to be used to monitor rotary parts, too. The electronic circuitry of the module delivers at its output in the time domain e.g. the maximum shaft vibration vector \( S_{max} \) (to VDI 2059, Sheet 3). An alarm is generated if a pre-set limit is reached. With the aid of the SUS computer, the number of conceivable false alarms can be reduced by:

- performance of plausibility analyses (check whether signal rise is present in all signals, whether only one of two signals for the same shaft motion rises, whether the amplitude increase is only temporary, etc.),
- making allowance for familiar phenomena during changes in operating parameters.

The module can also generate a prior warning alarm with the aim of increasing the time available between generation of the alarm and attainment of a critical equipment status to permit initiation of further analyses.

In addition, the SUS-86 presents frequency-selective information (e.g. difference spectra for shaft displacement amplitudes - Fig. 17), which can be used for generating an early stage alarm. This extends the early warning time still further.

*Fig. 17: Difference Spectrum PSD 1 – PSD 2 (monitoring of discrete amplitudes)*
6. CLOSING REMARKS

Ten years ago the eight functions involved in vibration monitoring were carried out in the Federal Republic of Germany only by experts using classic tools (Figure 18). The current general course of development is aimed at providing the user with immediate, technically unrestricted access to expert knowledge. The SUS-86 system for vibration monitoring constitutes a major step in this direction. It provides the user on the basis of the selected system parameters with unequivocal information as to whether the vibration deviations detected are tolerable or not.

![Activity chart]

In the course of progress towards the "expert system", the "deduction" activity is also to be integrated into the SUS-86 system in the next few years. At present, SUS-86 establishes the connection between deviations in vibration behaviour and affected components. Further applications and refinements are expected to stem from an R & D project supported by the West German government involving Kraftwerk Union AG and the Gesellschaft für Reactorsicherheit during 1984 to 1986. Among other things, the project used computer models to investigate the effects of postulated damage on monitoring signals /7/. Our ultimate goal is to prepare graded evaluation criteria which will show the operator which components must be replaced or repaired at which time. Whether and when this will be possible depends, above all, on what means will be available for this task in the years to come.

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DETECTION OF CORE BARREL MOTION AT WWER-440-TYPE REACTORS

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Abstract - A new method for detecting core barrel motion at WWER-440-type reactors by a pure excore detector arrangement is demonstrated. The method is based on excore neutron noise signals from different azimuthal positions and at least one acoustic sensor at the RPV outer surface that measures the sound generated by the leakage flow from the coolant inlet directly to the outlet through the labyrinth seal. If the core barrel vibrates there will be significant coherence between the envelope of the acoustic signal and any neutron noise signal of a chamber positioned at an angle ± 90° distant from the acoustic sensor. A pure core basket motion would not generate coherence. Though only core barrel displacement is mapped in excore neutron detectors, the amount of coherence between neutron noise signals depends on the position of the detectors and on the type of the core barrel pendulum motion. As well stochastic amplitudes in different directions as changes of the rotational sense decrease coherence. A mathematical model was developed to identify different types of core barrel motion.

1. INTRODUCTION

There is a great number of publications dealing with the detection of the vibration of reactor pressure vessel (RPV) internals (Fry, 1974; Bernard, 1982; Sweeney, 1986; Bastl, 1980 and Grabner, 1977). As it concerns core barrel motion the investigations by Fry, Kryter and Robinson (Fry, 1974) at the Palisades reactor became the classic example. They focussed attention to the mapping of core barrel displacements in the noise part of excore ionization chamber signals.

Up to now core barrel motion at WWER-440-type reactors has not yet been reported on. Since here the core barrel is fixed by 8 guide lugs near to the RPV bottom (see Fig. 1) such a vibration was expected to be improbable. In spite of this ionization chamber signals were measured at a WWER-440 that gave rise to suppose core barrel motion. The construction of the WWER-440—especially the fact that the core is shrouded in a so called core basket being capable of vibrating relatively to the core barrel (see Fig. 1)—renders the unambiguous detection of core barrel motion by means of excore ionization chambers more difficult. Therefore more detailed investigation was necessary to clarify which signals provide unique information.

2. DETECTOR ARRANGEMENT

Fig. 1 shows the most important internals of the WWER-440 RPV schematically and gives the detector arrangement which was used to obtain the results represented in ch. 3. Three pairs of ionization chambers were installed in 3 different channels azimuthally distributed round the RPV at positions about 120° distant from each other. The two chambers of each pair were in one channel, the upper (u) 2 m above the lower (l) one. Chambers J04u, J12u and J19u were near to the upper core edge, chambers J04l, J12l and J19l were near to the lower core edge.

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3. RESULTS

3.1. Neutron noise

3.1.1. Time domain and amplitude probability densities. In the left part of Fig. 2 the signals J121 and J191 are compared with a normal signal from a different reactor. All signals are normalized by their mean values. J121 and J191 are 7-8 times larger than the normal signal and so are all other neutron noise signals from the unit where core barrel motion was supposed.

Further J121 and J191 have a quite different structure in time than the normal signal. Particularly they are asymmetric. As can be seen from the right part of Fig. 2 normally a symmetric amplitude probability density is obtained and the peak values are in the range of \(+5 \times 10^{-5}\). To the contrary peak values of \(+4 \times 10^{-2}\) are registered for J121 and J191 at the unit under special investigation. Moreover Fig. 2 shows the skewness of the probability densities to depend on the azimuthal position of the ionization chamber. The probability for positive amplitudes is higher at J121 than for negatives and vice versa at J191. This hints to the fact that the force restoring the corresponding component to its rest position depends non-linearly on the displacement.
Detection of core barrel motion

3.1.2. Rms-values and power spectral densities. The power spectral densities of all 6 neutron noise signals are shown in Fig. 3 in the frequency range up to 8 Hz. There are no remarkable resonances, what is typical for chaotic vibrations of nonlinear systems (Troger, 1982; Schiehlen, 1987). Only at about 0.9 Hz a slight peak is to be seen especially in J121, J12u.

A comparison of the rms-values from upper and lower ionization chambers (Table 1, Table 2) in the frequency bands 0-1 Hz and 1-5 Hz provides more information about the vibration mode. In both frequency bands just the lower chambers give higher rms-values. The rms-ratio $\delta_{1}/\delta_{y}$ is in the range $6_{1}/6_{y}=1.21...1.26$. A pendulum mode vibration of the core barrel or of the core barrel internals (core basket and guide tube unit, Fig. 1) with the pivot at the core barrel support ledge or near to it would generate a geometric amplitude ratio $a_{y}/a_{u} = 1.35$. Obviously the difference between $6_{1}/6_{y}$ and $a_{y}/a_{u}$ mainly results from the overlapping detection fields of ionization chambers in one vertical channel.

Fig. 2. Time domain signals and amplitude probability densities of excore neutron noise during normal and core barrel motion conditions

Fig. 3. Power spectral densities of excore neutron noise signals from upper (u) and lower (l) ionization chambers
Table 1. Squares of the neutron noise rms-values $\sigma^2$, frequency band 0-1 Hz

<table>
<thead>
<tr>
<th>Detector</th>
<th>J04</th>
<th>J12</th>
<th>J19</th>
</tr>
</thead>
<tbody>
<tr>
<td>$\sigma_u^2$</td>
<td>$2 \cdot 10^{-4}$</td>
<td>$1.75 \cdot 10^{-4}$</td>
<td>$1.27 \cdot 10^{-4}$</td>
</tr>
<tr>
<td>$\sigma_{\perp}^2$</td>
<td>$3 \cdot 10^{-4}$</td>
<td>$2.77 \cdot 10^{-4}$</td>
<td>$1.96 \cdot 10^{-4}$</td>
</tr>
<tr>
<td>rms-ratio $\sigma_{\perp}/\sigma_u$</td>
<td>1.22</td>
<td>1.26</td>
<td>1.24</td>
</tr>
</tbody>
</table>

Table 2. Squares of the neutron noise rms-values $\sigma^2$, frequency band 1-5 Hz

<table>
<thead>
<tr>
<th>Detector</th>
<th>J04</th>
<th>J12</th>
<th>J19</th>
</tr>
</thead>
<tbody>
<tr>
<td>$\sigma_u^2$</td>
<td>$6.8 \cdot 10^{-5}$</td>
<td>$9.1 \cdot 10^{-5}$</td>
<td>$7.8 \cdot 10^{-5}$</td>
</tr>
<tr>
<td>$\sigma_{\perp}^2$</td>
<td>$1 \cdot 10^{-4}$</td>
<td>$1.43 \cdot 10^{-4}$</td>
<td>$1.2 \cdot 10^{-4}$</td>
</tr>
<tr>
<td>rms-ratio $\sigma_{\perp}/\sigma_u$</td>
<td>1.21</td>
<td>1.25</td>
<td>1.24</td>
</tr>
</tbody>
</table>

3.1.3. Signals from the same vertical channel (coherence and phases). Signals from the same string exhibit high coherence with no significant phase shift up to 4 Hz at least (see Fig. 4). This result confirms the assumption of a pendulum motion with the suspension above the upper chamber position.

![Fig. 4. Coherences and phases of excore neutron noise signals from chambers in the same vertical channel](image)

3.1.4. Signals from different azimuthal positions, frequency band 0.9 - 2 Hz (coherences, phases and trajectories). The coherences between ionization chamber signals from different azimuthal positions obtained at that special WWER-440-type reactor ($\gamma_{\text{max}} \approx 0.7$) are significantly smaller than those ($\gamma \approx 0.98$) known from the Palisades report (Fry, 1974). As it is shown in Fig. 5 there are two different frequency bands for coherence and phase, namely 0 - 0.9 Hz and 0.9 - at least 2 Hz. First the discussion of the higher frequency band 0.9 - 2 Hz.

The phases between signals in the same horizontal plane correspond to the geometric angle $\gamma$ the ionization chamber positions mark with respect to the reactor centre. Fig. 5 demonstrates the situation for the upper measuring plane.
If the trajectory of the RPV internals were elliptical the resulting phase shifts would be a function of both the positions of the detectors and the parameters of the ellipse. Fig. 6 shows some selected trajectories constructed from neutron noise assuming the fluctuations to be due to neutron transmission only. Each trajectory lasts about 3 secs. Reactivity effects can be neglected for external detector positions at least above 0.9 Hz. This is also confirmed by the phase relations themselves. As can be seen from Fig. 6 the trajectories are not circular what would explain the measured phases. The trajectories constructed from different signals agree very well for all three possible azimuthal pairs in one plane. Therefore the vibration of one internal component can be supposed to predominate. Always nearly, the trajectories are covered in the mathematical positive sense (B: begin, E: end), such segments excepted where the curves are strongly bended as the displacement of the concerning component were mechanically limited. Such segments occur seldom and will therefore mainly influence the low-frequency band below 0.9 Hz.

After frequency selective averaging at 1.22 Hz the mean trajectories depicted in Fig. 7 are obtained. They are almost circular with constant rotational sense.
3.1.5. Mathematical model. Deducing from the information given above the vibration of the internal component in the frequency band 0.9 - 2 Hz can be described by the following model:

- two dimensional stochastic pendulum motion without any preferred direction but constant rotational sense
- displacements from the rest position are approximately of the same amplitude for all directions and vary stochastically between 0 and any maximum value A

Starting from an elliptical trajectory with the displacements
\[ x(t) = a \cdot \cos \omega t, \quad y(t) = b \cdot \sin \omega t \tag{1} \]

in Cartesian coordinates \(x, y\) and measuring directions \(x', y'\) forming the angles \(\vartheta\) and \(\vartheta + \varphi\) with the direction \(x\) of the reference system (\(\varphi\) is the angle between two detectors) the above mentioned model is mathematically obtained by integrating \(S_x'(x', \omega), S_y'(y', \omega)\) and \(S_x(x, \omega), S_y(y, \omega)\) over \(\vartheta\) and over \(a, b\). Presuming uniform probability distributions \(0 < \vartheta < 2\pi; \ a, b \geq 0\) and statistical independence of all variables one gets
\[ \gamma(\omega) = \arctan(\frac{3/4}{\tan \varphi}) \tag{2} \]
\[ \gamma^2(\omega) = \cos^2 \varphi + \frac{9}{16} \sin^2 \varphi \tag{3} \]

\(\gamma(\omega)\) is the theoretical phaselseft between two ionization chamber signals, \(\gamma^2(\omega)\) is the maximum possible coherence for those signals. It is important to note that formula (2) and (3) only take into account varying neutron transmission through a water gap but not any possible reactivity effect. \(\gamma^2(\omega)\) according to (2) and (3) are depicted in Fig. 5 as dashed lines. Since \(\varphi\) is near to \(+120^\circ\) the difference between \(\varphi\) and \(\gamma\) is rather small.

The fact that the experimental coherence \(\gamma^2(\omega)\) really reaches the maximum possible coherence \(\gamma^2(\omega)\) with great probability proves the assumption of only one predominating vibration process inside the RPV. The simple model shows the coherences to be significantly lower than \(\gamma^2 = 1\) only due to the random nature of vibration.

3.1.6. Signals from different azimuthal positions, low-frequency band 0 - 0.9 Hz (coherences and phases). Coherences and phases in the low-frequency band below 0.9 Hz of Fig. 5 can be explained by slightly modifying the model (2), (3) of the higher frequency band. The requirement for constant rotational sense is given up. Mathematically this means, one of the amplitudes \(a\) or \(b\) (1) is uniformly distributed over \([-A, +A]\). Thereby one gets
\[ \gamma(\omega) = \begin{cases} 0^\circ & \text{for } \varphi < 90^\circ \\ 180^\circ & \text{for } \varphi > 90^\circ \end{cases} \tag{4} \]
\[ \gamma^2(\omega) = \cos^2 \varphi \tag{5} \]
The theoretical values according to (4) and (5) agree very well with the experimental values (see Fig. 5). Since $\phi > 90^\circ$, the phases are near to $+\pi$. The signal combination J12u - J04u excepted even the coherences from (5) envelop the measured curve. The experimental coherence J12u - J04u is higher than predicted by theory. As can be taken from Table 1, the low-frequency rms-value is significantly higher at positions J12 and J04 than at position J19. That fact and the clear $180^\circ$-phase shift hint at a preferred direction of the low-frequency pendulum motion from J04 to J12. This preferred direction of course is not considered in the model.

3.2. Coherences and phases between neutron noise and the acoustic signal at the labyrinth seal

The signal B06 is used to decide whether the core basket vibrates relatively to the core barrel or the core barrel together with the core basket relatively to the RPV. The acoustic signal B06 (see Fig. 1) is particularly generated by the leakage flow through the labyrinth seal directly from coolant inlet to outlet. The acoustic flow noise varies when the width of the gap in the seal and thereby the leakage flow is changed by core barrel motion. Coherence between the ionization chamber signals and the envelope $e(t)$ of the acoustic flow noise will exist only for core barrel motion but not for pure basket motion.

Fig. 8 shows coherences and phases between the envelope $e(t)$ and neutron noise signals. In the frequency band 0.9 - 2 Hz even here the experimental phase shift $\psi(\omega)$ corresponds with $\psi(\omega)$ from formula (2) marked as dashed line. The $180^\circ$-phase shift due to the different measuring effects (neutron transmission and leakage flow) has been taken into account when extracting the envelope. According to (4) below 0.9 Hz phase shifts of 0° or $180^\circ$ are obtained only. Remarkably even the coherences can at least qualitatively be described by (5) in the low-frequency band. The highest coherence results for the combination e-J19($\phi=30^\circ$), the lowest for e-J04 ($\phi=90^\circ$).

![Fig. 8. Coherences and phases between excor neutron noise and the envelope of the acoustic flow noise](image)

4. CONCLUSION

Core barrel motion can uniquely be detected in WNER-440-type reactors by proving coherence between excor neutron noise and the envelope of the acoustic signal due to the leakage flow through the labyrinth seal.

A comparison of experimental and theoretical coherences and phases of excor neutron noise signals from different azimuthal positions permits the conclusion whether an additional noise process contributes to the excor signals or not. The results of noise diagnostics have completely been confirmed by inspection. Based on these results an efficient method for core barrel motion detection with a pure external detector arrangement has been elaborated.

One should remark that neutron incore detectors also provide higher rms-values during core barrel vibration. The special reason for that could be clarified. Concerning this topic a special publication is being prepared (Schmitt and Weiß, 1988).

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REFERENCES


THIMBLE VIBRATION ANALYSIS AND MONITORING ON 1300 AND 900 MW REACTORS USING ACCELEROMETERS AND IN CORE NEUTRON NOISE

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ABSTRACT

The axial flow along the thimbles of the in core instrumentation induces vibration and shocks against their guides in the vessel, producing wear and even leakage, either on the thimbles, or on the instrumentation tube of the fuel assemblies. The new 1300 MW reactors were mainly concerned by this problem.

In order to characterize the phenomenon and help to reduce or suppress vibration of the thimbles, two methods have been developed and applied to French and Belgian reactors.

The first one consists of an analysis of the shocks perceived on the thimbles tubes by accelerometers; this analysis, based on the study of statistical distribution (amplitude, impulse rate of shocks...) has allowed to choose among the different solutions proposed to solve the problem; this choice has been confirmed by direct wear measurements made later.

The second method is based on spectral and time analysis of the fluctuating signals from in core neutron chambers. The correlation appears clearly between shocks and fluctuations. An estimation of the thimble modal shape in the instrumentation tube of the assembly, has been made.

These two analysis methods have been widely applied during start-up of the first eight 1300 MW reactors; they have contributed to solve the problem and to increase the availability of these plants.

On the 900 MW reactors, where the problem is less severe, the approach has been to study the mechanical behaviour of one new plant, Chupon B3: all in core guide tubes have been equipped with accelerometers and an on line monitoring system directly transmits to Chatou the parameters of shocks, in order to define an acoustic parameter able to characterize wear, and so, to define a new type of maintenance for the thimbles. The first results are presented.

MOTS CLEFS:

Doigt de gant - Instrumentation interne - Vibration - Choc - Bruit neutronique - Accéléromètre - Surveillance en ligne - SATIN - Usure par choc.
INTRODUCTION

Cette communication présente l'apport des techniques d'analyse de bruit à la résolution des problèmes vibratoires des doigts de gant de l'instrumentation interne sur les réacteurs 1300 et 900 MW.

Elle décrit deux méthodes, qui ont permis de caractériser les vibrations des doigts de gant sur le palier 1300 MW, et d'accélérer la résolution de ce problème vibratoire.

Elle présente également le système SATIN de surveillance en ligne des 50 doigts de gant d'un réacteur 900 MW (Chinon B3) ; SATIN a pour but d'évaluer la cinétique d'usure de ces doigts de gant, et de mettre en place une nouvelle technique de maintenance.

1 - PRESENTATION DU PROBLÈME

Première détection des chocs sur les réacteurs 900 MW

Tous les réacteurs français sont équipés d'un système de surveillance des corps errants et des vibrations des structures internes de la cuve (Réf. 1).

Dans le but d'assurer cette double surveillance au niveau de la cuve, trois accéléromètres sont montés en permanence sur les tube-guides (Fig. 1).

Pendant les essais précritiques à froid de la première tranche 900 MW, en 1977, les signaux délivrés par ces capteurs présentaient des impulsions quasi-périodiques. Ce phénomène existe sur l'ensemble des tranches 900 MW et nous l'avions expliqué par la présence de chocs des doigts de gant (ddg) contre leurs divers guidages dans la cuve.

Mais ces chocs ne nuisaient apparemment pas au bon fonctionnement du système de mesure de flux in core.

Le niveau des chocs était néanmoins suivi tous les jours à partir de ces accéléromètres de surveillance, et le phénomène était considéré comme normal, mais cependant nuisible à une bonne détection des corps errants en fond de cuve.

Première détection des chocs sur les réacteurs 1300 MW

En ce qui concerne les réacteurs 1300 MW, les chocs de ddg sont apparus dès le démarrage de la première tranche (Paluel 1) ; tous les signaux délivrés par les capteurs étaient perturbés par des chocs d'amplitude très élevée.

De même que sur les réacteurs 900 MW, une surveillance journalière de ces bruits était effectuée.

Or, après quelques mois de fonctionnement, deux fuites sont apparues sur les ddg des réacteurs de Paluel 1 et 2.

Explication du phénomène

Dans le but d'interpréter ce problème vibratoire, EDF a étudié le phénomène sur maquette et sur réacteur. Les essais ont mis en cause, comme source d'excitation, le flux axial le long du ddg. En effet, la vitesse du flux axial était plus élevée sur les réacteurs 1300 MW que sur les réacteurs 900 MW, ce qui a permis d'expliquer la différence de comportement vibratoire des deux familles de réacteurs.

Les ddg du palier 1300 MW vibraient donc sous l'effet du fort écoulement axial. Par suite, ils pouvaient entrer en collision avec leurs divers guidages internes à la cuve, à différents niveaux :

- sous le fond support, induisant donc une usure, voire une perforation du ddg ;

- dans le tube d'instrumentation, d'où une usure, voire un percement du tube d'instrumentation, de dureté et d'épaisseur plus faibles que le ddg.

Dans le but de réduire le niveau vibratoire et si possible de supprimer les chocs de ddg, plusieurs pièces mécaniques (appelées prothèses) ont été conçues, puis montées pour contrôler sur réacteur. De manière à juger de l'efficacité de ces prothèses, nous avons développé deux méthodes de mesures basées sur l'analyse de bruit ; la première utilise des accéléromètres, la deuxième les chambres neutroniques en coeur.
En ce qui concerne les réacteurs 900 MW où le problème vibratoire est moindre, un système (SATIN) de surveillance en ligne des doigts de gant, à partir d'accéléromètres, est installé à Chinon B3 et transmet quotidiennement des données sur Chatou.

2 - MESURE DES CHOCS DE DDG PAR ACCELEROMETRIE

Description
La méthode consiste à évaluer les amplitudes des impulsions et les intervalles de temps entre deux impulsions successives, contenues dans les signaux d'accéléromètres montés sur les prolongateurs de tube-guide, en salle d'instrumentation (fig. 1).

Les chocs entre le ddg et le guidage interne à la cuve génèrent des ondes qui se propagent le long du ddg, se transmettent au tube-guide sous la cuve, et se propagent le long de ce dernier jusqu'au capteur.

Intérêt de la méthode : caractérisation de l'état vibratoire du RIC d'une tranche donnée
La méthode permet donc d'identifier la présence de chocs et de connaître leur sévérité.

De plus, un essai particulier, effectué sur Paluel 3, a montré qu'il existe une bonne corrélation entre amplitudes de chocs et profondeur maximum de l'usure sur quatre ddg, après six semaines de fonctionnement de la tranche ; il faut d'ailleurs signaler à ce sujet que le ddg dont le niveau de choc était le plus élevé, était usé en quatre points, au lieu d'un seul pour les trois autres.

La méthode s'avère donc être un bon outil d'évaluation de l'endommagement des ddg : c'est sur son principe qu'est basé le système SATIN décrit dans le dernier paragraphe. De plus, il est envisagé de réaliser une maquette de reproduction du mécanisme d'usure, qui devrait donner accès à l'excitation réelle au niveau du point de contact.

L'application de la méthode accélérométrique sur les réacteurs 1300 MW a permis :

- de comparer les niveaux de chocs obtenus avec huit prothèses différentes, sur un même réacteur, lors des essais précritiques (Flamanville 1), et donc de sélectionner, puis monter, les meilleures prothèses sur les autres réacteurs ;

- de contrôler le bon comportement des prothèses sélectionnées et montées sur les réacteurs suivants.

La figure 2 compare les niveaux de chocs avant modification, puis après, lorsque la prothèse est en place : l'amplitude des chocs a été réduite d'un facteur 6 à 12, excepté pour Paluel 1 et 2, qui sont les seules tranches ayant fonctionné plusieurs mois (au cours de leur premier cycle) sans prothèse ; sur ces deux dernières tranches les histogrammes de répartition d'amplitude des chocs montrent que la prothèse ne semble efficace que sur 25 % des ddg environ.

La méthode d'analyse des fluctuations neutroniques permet d'expliquer le manque de performance de la prothèse.

3 - MESURE DES VIBRATIONS DE DDG PAR CHAMBRES NEUTRONIQUES EN COEUR SUR LES REACTEURS 1300 MW

Cette seconde méthode de mesure est basée sur l'analyse des fluctuations délivrées par les chambres en cœur à différents niveaux dans le coeur. Elle est utilisée par ailleurs, pour la surveillance vibratoire du combustible (Réf. 1).

Etat des réacteurs 1300 MW (et Doel 4) avant modification

Lors du démarrage de Paluel 1, des analyses neutroniques avaient révélé des spectres anormaux (fig. 3) : les pics liés aux vibrations normales de l'assemblage combustible et des internes étaient masqués par une multitude de pics, dus à des chocs répétitifs des ddg sur les tubes d'instrumentation de l'assemblage. La méthode accélérométrique a montré que tous les ddg, sans exception, étaient soumis à des chocs.

Nos collègues belges ont constaté le même phénomène à Doel 4.
État des réacteurs 1 300 MW après modification

Après l'apparition des fuites sur Paluel 1 et 2 et la mise en place de prothèses sur ces tranches, la forme des spectres s'est modifiée.

Comportement normal du dâg : sur 25 % des dâg de Paluel 1 - 2, le spectre est devenu normal et a fait apparaître les deux pics relatifs aux modes de l'assemblage combustible (Fig. 4) ; dans ce cas, le dâg vibre donc peu ou pas du tout, et l'accéléromètre correspondant sous la cuve ne détecte aucun choc ; sur ces dâg, la prothèse était donc inefficace.

Détection de vibration du dâg

Détection de la fréquence de vibration : sur les 75 % de dâg restants, un pic apparaît dans le spectre, aux environs de 14 Hz (Fig. 5). Il est dû au mouvement du dâg dans le gradient de flux ; cette fréquence coïncide avec le taux d'impacts détectés par l'accéléromètre correspondant. Aussi, dans ce cas précis, le dâg continue de vibrer ; la prothèse semble donc inefficace.

Détection de fuite du tube d'instrumentation : sur environ 40 % des dâg, d'autres pics apparaissaient, aux alentours de 20 à 35 Hz (bas de la Fig. 6), quand la chambre était en pied d'assemblage. Les pics n'existaient que sur les tranches irradiées (Paluel 1 et 2).

Ces deux modifications s'expliquent de la manière suivante :

Pendant les huit mois de fonctionnement antérieur de Paluel 1 sans prothèse, les dâg entraînaient en permanence en collision avec les tubes d'instrumentation, lesquels, d'épaisseur et de dureté plus faibles, s'usaient plus rapidement, certains allant même jusqu'au percement.

Du fait de l'existence d'une surpression à l'intérieur du tube d'instrumentation percé, un écoulement subsistait le long du dâg, provoquant sa mise en vibration à une fréquence de 14 Hz correspondant à environ 14 chocs/seconde ; des chocs se produisaient, comme auparavant ; ceci explique pourquoi la prothèse n'était pas efficace.

De plus, il existait un jet d'eau à travers la perforation, qui, par là-même, excitéait les crayons combustibles proches, les faisant vibrer aux alentours de 20 à 35 Hz ; cela est à l'origine de l'existence des autres pics du spectre.

Ce dernier phénomène est très voisin du jet de cloisonnement (Fig. 6). Cette analogie a permis de déterminer dès le redémarrage, l'existence d'une fuite des tubes d'instrumentation, et d'expliquer l'origine du pic à 14 Hz. Cette hypothèse de perçage a été confirmée par l'inspection d'un assemblage combustible qui avait été instrumenté pendant les huit premiers mois de fonctionnement.

Nous avons procédé à des études à différents niveaux dans l'assemblage. Ces investigations montrent une bonne corrélation entre les perturbations temporelles et fréquentielles des signaux neutrophiques, ainsi qu'entre signaux accélérométrique et neutrophonique (Fig. 7).

L'analyse des signatures neutrophiques à différents niveaux dans l'assemblage a permis d'évaluer la forme modale du dâg dans l'assemblage.

Forme modale du dâg : dans le but de connaître le déplacement réel du dâg dans le tube d'instrumentation, nous avons calculé et tracé le déplacement efficace de huit dâg (Fig. 8) ; ce calcul tient compte du gradient de flux dans le coeur.

Nous avons effectué ces calculs à partir des valeurs efficaces des pics sur les spectres ; deux, trois ou quatre points fixes, ou "noeuds", apparaissent sur le dâg, dans l'assemblage.

Les quatre comportements d'un dâg

Les analyses neutrophiques nous permettent de connaître le comportement de tout dâg (900 ou 1300 MW), que l'on peut ranger dans un des quatre cas suivants (Fig. 9) :

Cas A (réacteurs 1300 MW avant modification, DOEL 4, Tihange 3 et quelques dâg sur Tihange 2) : le dâg présente des chocs à différents niveaux, et l'accéléromètre correspondant perçoit également ces chocs.
Cas B et C (PALUEL 1 et 2 après modification):
le ddg vibre dans le tube d'instrumentation (premier pic) ; dans le cas B, le deuxième pic est dû à une excitation des assemblages voisins par jet transverse.

Cas D
(autres réacteurs 900 et 1300 MW, DOEL 3, TIXANGNE 2) :
il y a peu ou pas de vibration du ddg dans l'assemblage ; mais il peut y avoir des chocs en fond de cuve, que détecte alors l'accéléromètre correspondant.

4 - SURVEILLANCE EN LIGNE DES CHOCs DE DOIGT DE GANT SUR LES REACTEURS 900 MW

Le problème des ddg RIC est loin d'être aussi critique sur les réacteurs 900 MW que sur les réacteurs 1300 MW avant modification.

Néanmoins, sous l'effet des chocs internes à la cuve, les ddg s'usent ; les exploitants effectuent des contrôles de cette usure par courant de Foucault, lors des arrêts pour rechargement, en vue de remplacer les ddg les plus usés.

En ce qui concerne les dernières tranches du palier 900 MW en cours de démarrage (Chinon B3 et B4), EDF a décidé de les instrumer de ddg plus épais, afin d'allonger leur durée de vie. Nous avons équipé Chinon B3 d'un système, SATIN, dont le rôle est défini dans le prochain paragraphe.

Objectif de l'installation de SATIN

Le système SATIN répond à deux objectifs :

- Suivi en ligne du comportement vibratoire des nouveaux ddg, par auscultation journalière des 50 voies de mesure et transmission des données vers notre laboratoire.

- Accès à la cinématique d'usure des ddg, par mise en évidence d'une corrélation entre dommage D et coefficient de sollicitation cumulé Spq (F).

$D$, le dommage, représente une profondeur d'usure ou un volume de matière enlevé, éventuellement pondéré par la forme ou la cote de l'usure. D ne sera connu qu'après mesures des usures par courant de Foucault.

$Spq (F) = \sum_{i=1}^{F} \sum_{j=1}^{n} k(Nij \cdot (Ai)^p \cdot (Ai)^q)

Nij (Ai) est le nombre de chocs d'amplitude Ai advenus lors de l'acquisition du jour j.

n est le nombre total de chocs lors de l'auscultation de 100 secondes du jour j.

F correspond à la dernière acquisition, et donc au dernier jour du cycle combustible.

p et q sont des paramètres ajustables sur le système.

k = 864, est un facteur multiplicatif, permettant d'extrapoler à la journée les mesures faites sur 100 sec.

La connaissance de la corrélation entre usure et chocs permettra de définir une meilleure politique de maintenance des ddg sur les réacteurs 900 MW.

Nous n'avons malheureusement pas eu l'état actuel, de relevés d'usure sur les ddg de Chinon B3.

Aussi, après un bref descriptif du système, présenterons-nous les premiers résultats qu'il a permis d'obtenir concernant le comportement vibratoire des ddg.

Descriptif du système

Le système SATIN contient plusieurs étages :

- **Acquisition** : 50 accéléromètres sont montés en salle d'instrumentation, sur les tube-guides ; un calculateur, placé à proximité de la salle de commande, acquiert, voie après voie sur une durée de 100 secondes, les amplitudes de chocs dépassant un seuil de détection fixé.

- **Transmission des données** : les données (amplitude de chocs et intervalles de temps entre chocs successifs) sont compressées, puis transmises automatiquement par ligne téléphonique et modems vers notre laboratoire de Chatou, où elles sont traitées.

Cette transmission permet de reconstituer à Chatou le signal issu de l'accéléromètre (amplitude crête et temps d'arrivée de l'impulsion) (fig. 10).

**Premiers résultats**

Classement des ddg par ordre de sollicitation. Le coefficient de sollicitation cumulé $S_{11} (F)$ permet de classer les ddg en trois groupes, $F$ correspondant à une période de six mois :

- groupe des ddg très sollicités
  \[ \frac{S_{11} (F)}{k.F} > 50 \]

- groupe des ddg sans choc
  \[ \frac{S_{11} (F)}{k.F} < 0,5 \]

- groupe des ddg restant (avec chocs intermittents et faibles).

Intervalle de temps entre chocs successifs d'une même rafale : les histogrammes d'intervalles de temps entre chocs, montrent que le phénomène est quasi-periodique : la période varie de 75 msec à 200 msec. (voir Fig. 11), d'un ddg à l'autre, du fait des longueurs et conditions aux limites différentes de chaque ddg.

Sur les histogrammes, un deuxième pic (demi harmonique) peut apparaître (à 145 msec sur la fig. 11) et s'explique :

- soit par l'excentrement du ddg dans son guidage : le ddg ne rebondit alors que d'un seul côté ;

- soit par l'élimination (non dépassement du seuil de détection) des amplitudes correspondant à des rebonds de même parité dans leur ordre d'apparition.

Corrélation entre nombre de chocs et amplitude : une corrélation entre l'amplitude maximum des chocs détectés, et le nombre total de chocs pendant une acquisition, a été mise en évidence.

La Figure 12 montre qu'une relation linéaire apparaît très clairement (coefficient de corrélation de 0,965).

Mais la pente de la droite de corrélation varie d'une voie à l'autre, parce qu'elle est directement liée à la plage d'amplitude des chocs auxquels le ddg est soumis.

**Distribution statistique des amplitudes de chocs** : les amplitudes de choc d'un ddg très sollicité suivent une loi de Galton, excepté les amplitudes faibles, car ces dernières englobent en plus d'amplitudes de chocs, des crétes de bruit de fond dépassant le seuil de détection (voir Fig. 13).

**CONCLUSIONS**

L'analyse des signaux liés aux chocs des doigts de gant de l'instrumentation interne (accéléromètre et bruit neutronique in core) a beaucoup apporté à la technologie et à la garantie de l'intégrité des structures des réacteurs :

- sur les réacteurs 1300 MW, elle a permis un gain de temps appréciable dans l'évaluation des solutions proposées, en évitant l'arrêt des réacteurs après quelques semaines, pour contrôle de l'usure réelle des doigts de gant ; ce gain a été estimé à 72 jours de fonctionnement d'un réacteur à pleine puissance (Réf. 2) ;

- sur les réacteurs 900 MW, le problème est moins aigu ; le système SATIN, installé à Chinon B3, surveille en temps réel les chocs des doigts de gant ; il devrait permettre de définir une méthode d'analyse de bruit visant à estimer l'usure progressive des doigts de gant, et à contribuer à une meilleure politique de remplacement de ces derniers.
REFERENCES


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FIGURES

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Fig. 1 : Détection de chocs entre doigt de gant et colonne de guidage à partir d'un accéléromètre externe

Fig. 2 : Niveau de chocs maximum (+) et moyen (.) obtenus avant et après modification sur les réacteurs 1300 MW

Fig. 3 : Signature d'une chambre in core sur un réacteur 1300 MW avant modification
Fig. 4 : Absence de vibration d'un doigt de gant dans l'assemblage (DSPN d'une chambre in core en bas d'assemblage)

Fig. 5 : Vibration du doigt de gant (DSPN d'une chambre in core en bas d'assemblage)

Fig. 6 : Analogie entre jet de cloisonnement sur un réacteur 900 MW "down flow" et excitation due au jet transverse sur un réacteur 1300 MW irradié.

Fig. 7 : Comparaison spectrale et temporelle de signaux in core et accélérométrique sur des doigts de gant avec ou sans choc (PALUEL 1 avec prothèse et tube percé)
Fig. 8 : Estimation de la déformée modale de 8 doigts de gant dans l'assemblage combustible (PALUEL 1 avec prothèse et tubes percés).

Fig. 9 : Classification des signatures de chambres in core - Diagnostic et localisation des vibrations de doigt de gant.
Fig. 10 : Reconstitution en ligne à Chatou du signal délivré par un capteur situé sur un tube en salle d'instrumentation de Chinon B3 (partie positive)

Fig. 11 : Histogramme d'intervalle de temps entre chocs successifs sur un doigt de gant

Fig. 12 : Corrélation entre nombre et amplitude de choc sur un doigt de gant et sur des durées de 100 s.

Fig. 13 : Les amplitudes de choc sur un doigt de gant suivent une loi de Galton
Abstract - During 1986, in-core neutron noise measurements were made in a Swedish BWR, which indicated that instrument tube vibrations occurred in several cases. The experience gained during the interpretation of the measurements led to the setting up of a simple phenomenological model of in-core vibrating BWR detector spectra which is elaborated in the present paper. The novelty of the model is the inclusion of a global resonance-like term and its first harmonic into the noise components, representing the noise due to the detector itself vibrating in a linear and curved flux, respectively. Numerical calculations and comparison with measurements show a good qualitative agreement, which partly justifies the model itself, partly serves as a means of interpreting the measured spectra.

Keywords - BWR noise, instrument tube vibrations, self powered neutron detectors (SPNDs), phenomenological model, spectra, correlations.

1. INTRODUCTION

In the summer of 1986, in-core neutron noise measurements were performed in Barsebäck 1, a 550 MW Swedish BWR, to study signals from neutron detectors. After measurement of the noise for different detector strings, a spectral evaluation of the measured data followed in a routine way, in which signals of detectors in the same string were taken in one group so that cross-spectra between detectors in the same string could be calculated.

The theory of BWR in-core noise for axially displaced detectors is well-developed (Wach and Kosaly, 1974), and it is well-known that the phase of the CPSD shows a quasi-linear frequency dependence, whereas its absolute value or even more the coherence shows a regular sink-peak structure. There is a strict periodicity in both functions which is determined by the reciprocal of the transport time of the void between the two detectors.

Evaluation of the measurements showed, however, that the spectra in many cases significantly deviated from the expected and above described "ideal" pattern (Åkerhielm et al, 1986). Such changes included deviations of the phase from linear in frequency domains that varied in extension for different strings; most often the phase tended towards zero in those domains; characteristic dips occurred in the absolute value of the CPSD; the sink-peak structure of the coherence has been changed, so that either the sink frequencies were shifted, or the sink structure itself was destroyed; finally, peaks occurred at certain frequencies in the noise APSDs.

One rather plausible explanation of the above changes was that the instrument tube itself was vibrating during the measurements, and the vibrating component was interfering with the local transport and the global reactivity terms of the noise. This assumption was indeed used during the interpretation of the measurements. The fact that the instrument tubes did vibrate was confirmed later through independent investigations.

Although the fact that the spectra were affected by the presence of instrument tube vibrations was obvious in cases of strong vibrations, it still remained mostly a hypothesis in cases of small changes in the spectral characteristics (in those cases e.g. there was no peak seen in the corresponding APSDs). Thus to find some further justification of this hypothesis, at least a simple theory of BWR in-core noise in case of detector string vibrations became necessary.
To some surprise, however, we found that, although several papers have been addressing the problem of instrument tube vibrations (Thie 1975, 1979; Behringer et al., 1977), there is no work available that deals with the description of the shape of BWR noise spectra as measured by vibrating detectors. So we set up a phenomenological model very much the same way as the known models of BWR in-core noise in vibration-free cases (Wach and Kosaly, 1974). The extension we made was to include a resonance-like "global" component besides the local transport and broad-band global reactivity noise terms. The frequency shape of this term was determined by the response of a damped harmonic oscillator driven by a white noise force. This was then converted to neutron noise through the assumption of a flux gradient. It is just as easy to calculate the effect of vibrations on the noise spectra if the curvature of the static flux is taken into account, in which case two resonances will appear in the APSD, the second one corresponding to the well-known double frequency effect.

Although it is fairly straightforward to describe and interpret analytically the above three components of the noise, it is somewhat more difficult to assess the result of the interference of all three components. To this end we evaluated numerically the analytical formulae, describing the model, at various values of its free parameters. By changing such parameters as the vibration frequency, transit time, global-local ratio, vibration amplitude, ratio of the second to the first derivative of the static neutron flux, etc., a relatively complete mapping of the structure of BWR noise with detector vibrations could be achieved. These calculated values were then plotted and compared against the measured data. By a suitable choice of model parameters, with very few exceptions, all characteristics of the measured data could be reconstructed to a high degree.

The good agreement between the measured and calculated data served primarily as a justification of the model itself. This in turn means, however, that the model calculations, since they refer to cases where the vibration properties are input parameters and are thus at least qualitatively known, can contribute to the understanding and interpretation of the measured spectra.

The paper consists of two parts. First the phenomenological model is defined and developed analytically. The different noise components are defined, and the spectra are calculated. In particular, the frequency dependence of the vibration term is calculated for the case where the motion takes place in a curved flux. The second part consists of a numerical evaluation of the auto and cross spectra (amplitude and phase), which are then compared against the measured spectra.

2. PRINCIPLES OF THE MODEL

In this report there is only room for a short description of both the model and the calculated and measured results. A more detailed account of the model and the agreement between theory and experiment is expected to be reported in a later communication.

As has already been mentioned, we set up a phenomenological model of the flux fluctuations as measured by a vibrating detector in a BWR core. We shall assume that a single detector signal in the frequency domain consists of the following three components:

\[ S(w) = L(w) + G(w) + V(w) \]  \hspace{1cm} (1)

where \( L(w) \), \( G(w) \) and \( V(w) \) stand for the local and the global components of the vibration-free BWR noise, whereas \( V(w) \) is the term that accounts for the effect of the vibrations. We shall assume that these three terms are uncorrelated, that is

\[ \langle L(w) G(w) \rangle = \langle G(w) V(w) \rangle = \langle L(w) V(w) \rangle = 0. \]  \hspace{1cm} (2)

The functional form of the above components is chosen as follows. Since we want to calculate correlations between detectors in different axial positions in the same string, axial dependence of the noise must be allowed for. Hence we write

\[ L(w) = A(w) \exp(-i\omega z/v). \]  \hspace{1cm} (3)

Here \( A(w) \) is a real function, proportional to the void fluctuations at the inlet point \( z=0 \). \( z \) itself is the axial position of the detector, and \( v \) is void velocity. The actual form of \( A(w) \) is

\[ A(w) = 1 / \left[ 1 + (w/w_c)^4 \right]. \]  \hspace{1cm} (4)
For the global reactivity term we assume a real function in the form

\[ G(w) = \left[ A(w) + C \right] / k. \]  

(5)

Here \( C \) is a real number whose value was kept constant during the calculations, whereas \( k \) is the local/global ratio introduced by Wach and Rosaly (1974).

The selection of real functions for \( A(w) \) and \( G(w) \) is motivated by the fact that we shall do our investigations around the plateau region of the neutron physical transfer of a commercial BWR, where the time delay in the detector signals is dominated by transport effects in the coolant over phase delays in the neutron physical transfer. The actual functional form in (4) and (5) was selected to resemble the measured auto spectra shapes as closely as possible.

Regarding the term \( V(w) \), we assume that it is due to the response of a detector vibrating in a static flux. For simplicity, we shall assume one-dimensional vibrations, but it is easily seen that accounting for the two-dimensionality of the realistic vibrations does not affect the frequency behaviour of the spectra. Then, we shall have for the detector signal in the time domain, up to the second order of the displacement amplitude

\[ V(t) = \gamma \left[ a_1 x(t) + a_2 x^2(t) \right] \]  

(6)

where \( a_1 \) and \( a_2 \) stand for the first and second derivatives of the static flux, respectively, and \( \gamma \) is the scaling factor between flux change and detector current. Then, from (6) one has

\[ V(w) = \gamma \left[ a_1 x(w) + a_2 y(w) \right] \]  

(7)

with (Williams, 1980)

\[ y(w) = \int_{\infty}^{\infty} \exp(-iw't) x^2(t) \, dt = 1/(2\pi) \int_{\infty}^{\infty} x(w') x(w-w') \, dw'. \]  

(8)

If we assume a damped forced vibration with a white noise driving force, then (Paidoussis, 1974)

\[ x(w) = 1 / \left[ (w^2-w_0^2) + 2i Dw \right] \]  

(9)

where \( w_0 \) is the eigenfrequency of the vibrating system and \( D \) is the damping factor. Here, without restricting generality, the amplitude of the force PSD was chosen as unity; but clearly, the amplitude of the force spectrum, whatever it differs from unity, can be included into e.g. the factors \( a_1 \) and \( a_2 \).

It remains now to calculate \( y(w) \) from (8) and (9). We have to evaluate

\[ y(w) = \frac{1}{2\pi} \int_{\infty}^{\infty} \left[ (w-w')^2 - w_0^2 \right]^2 + 2iD(w-w') [(w'^2-w_0^2) + 2i Dw']^{-1}. \]  

(10)

This integral can best be evaluated by the theorem of residues if one notices that the integrand vanishes as \( 1/|w| \) as \( |w|=|u+iv|=\infty \). There are two poles in each half plane with equal sum of residues, and it does not matter which way we close the contour. The result is

\[ y(w) = 2i/\left( (w+2iD) [(w^2-4w_0^2) + 2i Dw] \right). \]  

(11)

It is seen that \( y(w) \) has a spectrum resonance at \( w = 2w_0 \), i.e. this term is responsible for the double frequency effect, as expected. Combining (6) – (11) gives finally

\[ V(w) = v_1 x(w) + \alpha v_2 y(w). \]  

(12)

Here the factors \( v_1 \) and \( v_2 \) are the product of the corresponding flux gradients, the \( \gamma \) scaling factor, and the appropriate power of the force PSD. Through this latter, they are related also to the vibration amplitude, although this relationship is rather implicit. \( \alpha = \pm 1 \) is a phase factor so that both \( v_1 \) and \( v_2 \) can always be taken as positive.
We now have all functions that are necessary to calculate detector auto and cross spectra. Due to the uncorrelated property of the noise components as defined in (2), for a detector APSD we have:

$$\text{APSD} = S_{LL}(w) + S_{GG}(w) + S_{VV}(w)$$

with $S_{LL}(w) = \lambda^2(w)$, $S_{GG}(w) = G^2(w) = \left[ \lambda^2(w) + C^2 \right] / \kappa^2$

and $S_{VV}(w) = |V(w)|^2 = v_1^2 |x(w)|^2 + v_2^2 |y(w)|^2 + 2 \alpha v_1 v_2 \text{Re}(x(w)y^*(w))$

where $|x(w)|^2$, $|y(w)|^2$ and $\text{Re}(x(w)y^*(w))$ are readily calculated from (9) and (11).

To calculate the cross spectra, we assume two detectors at positions $z_1$ and $z_2$. Then we have for the cross spectra

$$\text{CPSD} = S_{LL}(w) + S_{GG}(w) + S_{VV}(w)$$

with $S_{LL}(w) = \lambda^2(w) \exp(-i\omega t)$, $\tau = (z_1 - z_2)/v$

$$S_{GG}(w) = [\lambda^2(w) + \text{corr} \cdot C^2] / \kappa^2$$

and $S_{VV}(w) = V_1(w) V_2^*(w)$

$$V_1(w) = v_1 x(w) + v_2 y(w); \quad V_2(w) = \text{fac} \cdot [v_1 x(w) + \beta v_2 y(w)]$$

Here again, $\text{fac} = \pm 1$, $\alpha = \pm 1$ and $\beta = \pm 1$ are phase factors defining the in-phase and out-of-phase behaviour of the different constituents of the vibration term. Negativity of any of these factors corresponds to a case when the two detectors move in opposite flux gradients or curvatures. The constant corr defines the correlated part of the background noise. The same notations were used in the figures shown below. From the above, the absolute value and the phase of the CPSD as well as the coherence can be calculated.

It is easily seen that without the vibration term $V(w)$ and apart from further minor differences, the model developed above shows considerable resemblance to the phenomenological model of Wach and Kosaly (1974). For example, taking $A(w) = G(w) = 1$ and $V(w) = 0$ leads to exactly the same periodic (and non-decaying) sink and peak structure and quasi-linear phase shown there. The present model is aimed partly at a better description of the frequency-dependent parameters of the model, and mostly at an assessment of the effects of the vibration term and its first harmonic that are incorporated in $V(w)$.

One remark is, however, regarding the absolute value of the model parameters. As seen from the formulae, the derived parameters $v_1$ and $v_2$ are not yet given a direct physical meaning. This is impossible until one knows exactly such things as the scaling between detector response for a unit displacement in unit flux gradient and that for the passage of a unit bubble, etc., not to mention the absolute values of the flux gradient and the variance of void fraction fluctuations. This means that a good agreement between measured and calculated results does not mean that one can determine the vibration properties on an absolute scale. The emphasis was on tendencies and on discovering features that can help developing diagnostic methods in a qualitative way.

3. NUMERICAL CALCULATIONS AND COMPARISON TO MEASUREMENTS

The formulae obtained above are a little cumbersome for an analytical evaluation of their features. A relatively simple method is, however, to evaluate them numerically for various values of the parameters that appear in the model. By varying the parameters that are related to vibration properties, notably $v_1$ and $v_2$, the effect of vibrations with different amplitudes on the spectral properties can be followed. As is clear from the foregoing, with taking different ratios of the $v_1$ and $v_2$ terms, one can also account for different flux buckling effects. Setting $v_1 = v_2 = 0$ corresponds to the vibration-free case.

Results are shown in a series of figures below. Since the main objective was to see if the model can reconstruct the characteristic spectral shapes that we found in the experiment, the figures are arranged in several sets. Each set consists of two parts. First, spectral descriptors are shown that belong to one particular measurement, and which were found to display some characteristics of the effect of vibrations. The second part of the set consists of the calculated data with one fixed set of model parameters. It is important to notice that when
varying the model parameters, a simultaneous agreement between all three spectral quantities (APSD, CPSS amplitude and CPSS phase) was sought for. This was obviously harder to achieve than to get agreement between single amplitude or phase pairs, but at the same time provides a deeper proof of the validity of the model.

Before discussing the figures, it might be worth listing what type of characteristics of the noise we wanted to reproduce. The characteristics of the vibration-free case are well understood (Wach and Kosaly, 1974) and were already described in the introduction. In the cases where vibrational effects are also present, the following characteristics were noticed in the measured data (Akerheim et al, 1986):
- appearance of characteristic sinks, sometimes quite deep, in the absolute value of the CPSS;
- a distortion of the phase behaviour over a part or the whole of the domain of linear behaviour;
- in case of stronger vibrations, appearance of one or two peaks in the APSD.

Figures 1 and 2 correspond to the vibration-free case. Up to approximately 12 Hz, the measured data display a sink structure and a rather linear phase with sink-peak frequencies and zero- and 180°-crossings of the phase completely in accordance with expectations. The sink structure is displayed by the coherence which is not shown here for brevity. The model calculations show a rather good agreement with the measurements, not only in terms of the sink structure, but also in the overall behaviour of the spectral quantities over the whole frequency range.

The measured data, shown in the following, display most of the characteristics that we found during the evaluation of the measurements and which were mentioned above. Based on several calculations with different model parameters that cannot all be shown here, it is claimed that all these characteristics of the BWR spectra affected by detector vibrations can be reconstructed. The fact that these characteristics could be reconstructed by the model supports the interpretation of the measured data in the above quoted report.

Figures 3 and 4 refer to a case of relatively weak vibration effects where by weak vibrations we mean that the magnitude of the vibration term was not dominating over the others in (1); as discussed earlier, this does not necessarily mean small vibrations. For weak vibrations it is reasonable to assume that the higher harmonics can be neglected.

Figure 3 displays the characteristics of the effect of weak vibrations on the measured spectra as discussed by Akerheim et al (1986), namely that if the slope of the APSD is relatively high and the vibration peak is broad, the effect of vibrations can hardly be seen on the APSD. The absolute value of the CPSS, and especially the phase are much better indications of weak vibrations.

On Figure 4 a series of calculations are shown where the absence of higher harmonics is assured by taking \( v = 0 \). The change of the spectra as compared to Figure 2 to contain the above characteristics can easily be observed.

Figure 5 shows "stronger" vibrations (as might be obvious, that can mean both larger amplitudes or/and higher flux gradients). This observation is based on the fact that the vibration peak in the APSD is now observable and the sinks in the CPSS are more pronounced. With the model calculations, shown in Figure 6, these features could be once again reconstructed. It is interesting to note, that to achieve this simultaneously for both auto and cross spectra (amplitude and phase), the presence of the \( v \) term had to be assumed too, and further its sign needs to be opposite to the \( v \) term for both detectors (this is indicated by the choice of \( \text{fac} = 1, \alpha = -1 \text{ and } \beta = -1 \) in Figure 6. It is interesting to notice that again, the presence of the higher harmonics is indicated by the distortion of the phase but is not seen in the APSD.

Figures 7-10 show two cases taken from different measurements. The last two figures (Figures 11 and 12) refer to a third case where the most violent vibrations were found in the evaluation of the measured data (Akerheim et al, 1986). Here the double frequency component can be seen even as a peak in the APSD.

4. CONCLUSIONS

The comparison of the measured and calculated data shows that the characteristics of the measured data can be reconstructed and thus can also be interpreted based on the model developed here. Certain qualitative features and even some quantitative parameters such as vibration frequency, damping factor, etc. can be found or at least estimated. Due to reasons described earlier, however, a quantitative statement about the absolute value of the magnitude of the vibrations is not possible to make based on the present model.
The possible use of the above reported findings can be summarized as follows. First, it was shown that the phase of the cross spectra is the most sensitive indicator of the vibrations. It is therefore seen that this parameter is a suitable feature to use in an early warning system to indicate instrument tube vibrations. Another extreme case that might be interesting in a vibration monitoring system is the case of impacting. Obviously, such effects are not included in the present model. We found that in this model, however, increasing the amplitude of the vibration and introducing non-linear transfer between displacement and neutron noise did not lead to a broadening of the vibration peaks in the APSD. Based on recent simulation work (Glöckler and Frei, 1987), it was suggested that peak broadening can be an indicator of impacting. Our investigations support this view by showing that double frequency effects do not cause peak broadening.

The above reported results suggest that it might be worth doing some further work along the same lines that could lead to the enhancement of the applicability of the method in practical diagnostic tasks. On the other hand, some results of the model might not be generally valid to all BWR plants; some further investigations could clarify this point as well.

ACKNOWLEDGEMENT
We are indebted to Dr. J.A. Thie for valuable discussions. Thanks are due to Dr. O. Glöckler and Z. Frei for discussions and for communicating to us their results prior to their publication.

REFERENCES
Thie, J.A. (1975) Nucl. Technology 21, 532
Wach, D. and G. Kosaly (1974) Atomkernenergie 23, 244
Fig. 1a. APSD, measurement

Fig. 1b. CPSD amplitude, measurement

Fig. 1c. CPSD phase, measurement

Fig. 2a. APSD, model

Fig. 2b. CPSD amplitude, model

Fig. 2c. CPSD phase, model

Fig. 1 and Fig. 2. Comparison between results from measurement and model. No vibration effects.
Fig. 3a. APSD, measurement

Fig. 3b. CPSD amplitude, measurement

Fig. 3c. CPSD phase, measurement

Fig. 4a. APSD, model

Fig. 4b. CPSD amplitude, model

Fig. 4c. CPSD phase, model

Fig. 3 and Fig. 4. Comparison between results from measurement and model. "Weak" vibration effects.
Fig. 5a. APSD, measurement

Fig. 5b. CPSD amplitude, measurement

Fig. 5c. CPSD phase, measurement

Fig. 6a. APSD, model

Fig. 6b. CPSD amplitude, model

Fig. 6c. CPSD phase, model

Fig. 5 and Fig. 6. Comparison between results from measurement and model. "Strong" vibration effects.
Fig. 7. CPSD phase, measurement

Fig. 8. CPSD phase, model

Fig. 9. CPSD phase, measurement

Fig. 10. CPSD phase, model

Fig. 11. APSD amplitude, measurement

Fig. 12. APSD amplitude, model
ANALYSIS OF INTERNALS VIBRATION MONITORING AND LOOSE PART MONITORING SYSTEMS DATA RELATED TO THE ST. LUCIE I THERMAL SHIELD FAILURE

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Combustion Engineering, Inc., Windsor, Connecticut 06095, U.S.A.

St. Lucie Unit 1 (Florida Power & Light) is a 890 MWe PWR which entered commercial operation in April, 1976. During the post cycle 5 refueling outage in March, 1983, the thermal shield was found to be damaged. The investigation into the cause of the failure considered a number of areas possibly related to the failure.

As part of this investigation, both internals vibration monitoring (IVM) and loose parts monitoring (LPM) data were reviewed and reanalyzed, using state of the art techniques, developed after discovery of the damage to determine changes in the dynamic characteristics of the core barrel-thermal shield system during its operating history.

The objective of this paper is to present results of the reanalyses of the internals vibration and loose parts monitoring systems data over the operating lifetime of the plant. The paper demonstrates how dynamic analysis of the core barrel-thermal shield system was utilized in the interpretation of the IVM and LPM data.

1. INTRODUCTION

In March of 1983, at the end of the fifth fuel cycle, visual inspection of the St. Lucie 1 reactor vessel internals disclosed that the thermal shield and its support system were damaged. An analysis of the mechanism that caused the damage was undertaken to provide an explanation of the cause of the failure.

The objective of this paper is to present the results of an evaluation of the IVM and LPM data done as part of this analysis.

2. CORE SUPPORT ASSEMBLY

The core support assembly consists of the core support barrel, the lower support structure, the core shroud, and the thermal shield (Fig. 1). The material for the assembly is Type 304 stainless steel.

The core support barrel is suspended by a flange from a ledge on the pressure vessel and supports the lower support structure upon which the fuel assemblies rest. The snubbers consist of six equally spaced double lugs around the circumference and are the grooves of a "tongue-and-groove" assembly.

The thermal shield is supported at the top by nine equally spaced support lugs welded to the periphery of the core support barrel. Support pins, welded to the thermal shield, are fitted during assembly to position the thermal shield on the support lugs. The thermal shield is positioned radially by a total of twenty-six positioning pins; nine located 38.1 cm (15 in.) below the top of the support lugs and the remaining located 53.9 cm (21.25 in.) from the bottom.

3. CHRONOLOGY

A chronology of the operating history, including visual inspections and analyses of loose parts and excore detector monitoring data, are shown in Table 1.

Baseline values on the loose parts monitoring (LPM) system lower head accelerometers were first exceeded in February, 1978. Analysis of LPM data concluded that the signals were due to
a small loose part free to move in the lower head of the reactor vessel. Remote visual inspection of accessible areas in 1978 failed to locate a loose part.

![Diagram of reactor core support barrel and thermal shield]

**FIGURE 1**
ST. LUCIE 1 CORE SUPPORT BARREL AND THERMAL SHIELD

<table>
<thead>
<tr>
<th>DATE</th>
<th>CORE</th>
<th>EVENT</th>
</tr>
</thead>
<tbody>
<tr>
<td>10-75</td>
<td>0</td>
<td>Calibration of LPM System</td>
</tr>
<tr>
<td>4-22-76</td>
<td>1</td>
<td>Critical IVM Baseline,</td>
</tr>
<tr>
<td>7-10-76</td>
<td>Outage</td>
<td></td>
</tr>
<tr>
<td>12-4-76</td>
<td>1A</td>
<td>Start, Cycle 1A</td>
</tr>
<tr>
<td>2-2-78</td>
<td>1A</td>
<td>LPM set point exceeded</td>
</tr>
<tr>
<td>3-28-78</td>
<td>Outage</td>
<td></td>
</tr>
<tr>
<td>5-26-78</td>
<td>2</td>
<td>End of Cycle 1A</td>
</tr>
<tr>
<td>6-2-78</td>
<td>2</td>
<td>Study of LPM signals</td>
</tr>
<tr>
<td>4-1-79</td>
<td>Outage</td>
<td></td>
</tr>
<tr>
<td>6-8-79</td>
<td>3</td>
<td>End, Cycle 2</td>
</tr>
<tr>
<td>3-11-80</td>
<td>3</td>
<td>Start, Cycle 3</td>
</tr>
<tr>
<td>3-15-80</td>
<td>Outage</td>
<td></td>
</tr>
<tr>
<td>5-7-80</td>
<td>4</td>
<td>Increase in LPM alarms</td>
</tr>
<tr>
<td>9-8-81</td>
<td>Outage</td>
<td></td>
</tr>
<tr>
<td>11-29-81</td>
<td>5</td>
<td>End, Cycle 3</td>
</tr>
<tr>
<td>9-25-82</td>
<td>5</td>
<td>Start, Cycle 5</td>
</tr>
<tr>
<td>3-4-83</td>
<td>Outage</td>
<td></td>
</tr>
<tr>
<td></td>
<td></td>
<td>Increase in LPM alarm rate</td>
</tr>
<tr>
<td></td>
<td></td>
<td>End, Cycle 5</td>
</tr>
</tbody>
</table>

**TABLE 1** CHRONOLOGY OF ST. LUCIE 1 OPERATION
In March, 1980, new alarm levels, set after the 1978 analysis, were exceeded. An evaluation concluded that the signals were not related to motion of the core support barrel. In 1981, remote visual inspection of accessible areas did not reveal damage or loose parts.

In September, 1982, an increase in the frequency and magnitude of the LPM signals resulted in a program to continually record and evaluate LPM system data. The evaluation indicated that the alarm rate and magnitude had stabilized, with no significant changes noted between September, 1982, and the shutdown in March of 1983, when damage to the shield was found.

A program was established to determine the cause of the failure. This paper reviews the results of the portion of the post damage program related to evaluation of the IVM and LPM data.

4.0 FREE VIBRATION OF CORE SUPPORT BARREL AND THERMAL SHIELD

The natural vibration characteristics of the coupled core support barrel and thermal shield system were analyzed to determine and quantify the change in vibration with assumed changes in the thermal shield support system. The results were then utilized in interpreting the IVM and LPM data.

To calculate frequencies and mode shapes, a three-dimensional finite element shell model was developed for use with the SAP4 computer code. The core support barrel, thermal shield, and support lugs were modeled with plate elements, having both membrane and bending capability. Details of the core support barrel such as the outlet nozzles and thickness change, were represented in the model. Boundary conditions were imposed at the interface between the support lug and the thermal shield to represent the as-built condition and the change in support as damage progressed. The model included the lower core support structure. The top of the core support barrel was considered fixed for all cases analyzed. Hydrodynamic coupling effects, due to fluid in the annuli on both sides of the shield, were considered in calculating the in-water frequencies.

Frequencies in air and water and mode shapes from the coupled core support barrel and thermal shield model were calculated for as-built conditions, with pins removed, and with support lugs removed. Results showed that only modes 1 through 4 contribute significant response to normal operating hydraulic loads. In a separate analysis, the thermal shield was removed from the model and the core barrel natural frequencies were calculated. The frequencies of the core barrel alone are close to those of the coupled system, indicating that the vibration is dominated by the core barrel unless thermal shield support components are removed from the model. A summary of cases is presented in Table 2.

<table>
<thead>
<tr>
<th>MODE</th>
<th>TYPE</th>
<th>NO POSITIONING</th>
<th>PINS</th>
<th>NO POSITIONING</th>
<th>PINS</th>
<th>CLAMPED</th>
<th>SUPPORTED</th>
</tr>
</thead>
<tbody>
<tr>
<td>1</td>
<td>BEAM</td>
<td>6.8</td>
<td>6.7</td>
<td>6.7</td>
<td>6.7</td>
<td>6.9</td>
<td>10</td>
</tr>
<tr>
<td>2</td>
<td>SHELL</td>
<td>7.6</td>
<td>5.1</td>
<td>3.1</td>
<td>8.2</td>
<td></td>
<td></td>
</tr>
<tr>
<td>3</td>
<td>SHELL</td>
<td>11.3</td>
<td>10.5</td>
<td>10.4</td>
<td>12.0</td>
<td></td>
<td></td>
</tr>
<tr>
<td>4</td>
<td>SHELL</td>
<td>18.9</td>
<td>17.7</td>
<td>17.5</td>
<td>17.2</td>
<td></td>
<td></td>
</tr>
</tbody>
</table>
For the case of no top pins, little change in the coupled frequencies and mode shapes was observed. However, removal of the bottom pins causes a significant change in the natural frequency and shape of the first shell mode (n=2) of the coupled system but has little effect on the remaining modes.

A significant change in the second mode of vibration was also observed when all but vertical restraint is removed from the lugs. The thermal shield becomes uncoupled from the core barrel and its natural frequency is reduced. These results show that absence of bottom pins, and lugs affects the first shell mode of vibration and reduces frequencies. Plots of deflected shapes for modes 1 and 2 are shown in Figure 2.

**FIGURE 2**

Modal response of core barrel and thermal shield
5.0 MONITORING DATA

Surveillance monitoring of reactor vessel internals was accomplished on St. Lucie 1 by a Loose Parts Monitoring System (LPM) and an Internals Vibration Monitoring (IVM) System, both furnished by Combustion Engineering, Inc. This section presents information on the results of analyses of the signals from both of these systems, performed after the discovery of damage to the thermal shield.

5.1 Loose Parts Monitoring (LPM)

The LPM system utilizes accelerometers mounted on the outside of the reactor vessel and steam generators (Figure 3) to record signals that could be caused by the impact of a loose part (1,2). Monitoring at St. Lucie 1 is done continuously, with the accelerometers set to provide alarms at preset values.

Three sets of magnetic tapes were recorded in conjunction with three previous studies (Table 1). A post-damage analysis of the LPM signals for these three tapes was done using a method developed during this investigation which treats the LPM signals in a stochastic manner (3,4). The modified amplitude probability distributions (MAPD) from the three magnetic types are shown in Figure 4. It is apparent that the magnitude of the impacts increased from 1978 to 1982. The peak g values are summarized in Table 3.
### TABLE 3 POST OUTAGE ANALYSIS LPM PEAK "G" VALUES

<table>
<thead>
<tr>
<th>LOWER HEAD, REACTOR VESSEL</th>
<th>UPPER HEAD, REACTOR VESSEL</th>
</tr>
</thead>
<tbody>
<tr>
<td>1</td>
<td>2</td>
</tr>
<tr>
<td>1978</td>
<td>1.0</td>
</tr>
<tr>
<td>1980</td>
<td>1.3</td>
</tr>
<tr>
<td>1982</td>
<td>1.9</td>
</tr>
</tbody>
</table>

* BACKGROUND - NO IMPULSES NOTED

The following observations were made: (1) There was an increase in peak g level on all channels with operating time. (2) All channels show about the same g level in 1980 but channels 3 and 4, at the top of the vessel, show a much greater increase from the previous levels. (3) The 1982 data shows a large increase in channels 3 and 4 and little change in channels 1 and 2.

### 5.2 Internals Vibration Monitoring (IVM)

The IVM system relies on the excore neutron detectors (Figure 5) to record random variations in neutron flux due to changes in neutron absorption path length caused by both neutronic effect and motion of the reactor internals (5,6). Monitoring at St. Lucie I was done at the beginning and end of each fuel cycle and at four month intervals during each cycle.

![Diagram of reactor internals and detector locations](image)

**FIGURE 5** ST. LUCIE I EXCORE DETECTOR LOCATIONS

With regard to core support barrel motion, a comparison of the RMS amplitudes in a frequency band about the 7.5 Hertz beam mode frequency and the wide band (2-20 Hertz) RMS values showed no significant changes in value (Figure 6).
The objective of the post-damage analysis of the IVM data was to identify changes in the noise spectrum with operating time which could be indicative of changes in the thermal shield support system, and to permit comparison of the results with calculated structural frequencies for assumed support conditions summarized in Table 2.

Post-damage, Phase Separation Analysis (7) was done of BOC and EOC data of Cycles 2 to 5. Results are summarized in Figure 7. The following observations can be made:

A. The data for cycles 2 and 3 are similar; the out of phase portion dominated by the peak associated with core support barrel motion (n=1 mode) at 7.5 Hertz, the in phase portion having dominant peaks at about 7.5 (n=2 mode) and 13 (n=3 mode) Hertz. The coherence for both sets of data is similar.

B. The coherence for cycle 4 shows a marked departure from that of cycles 2 and 3, having a predominantly four peak character; the peak from 2 to 11 Hertz appearing as two, one from 2 to 5 and another from 5 to 11 Hertz.

Inspection of the phase separated PSDs shows that this change was due to a shifting of the in phase n=2 peak at 7.5 Hertz to a lower frequency of about 5 Hertz. These changes correspond to predicted reductions in shell mode (Table 2) with assumed changes in support conditions.

C. A small peak was evident at about 10 Hertz in the out of phase data.

This peak could correspond to the thermal shield beam mode calculated for the removal of all positioning pins. There were no changes in the peaks at 13 and 19 Hertz. This corresponds well to calculations for the remaining shell modes.

D. The cycle 5 data shows a reduction in frequency of the peak originally at 5 hertz with operating time.

A review of the PSDs and the coherence shows that this change was due to further reductions in the inphase peak at about 5 Hertz to about 4 Hertz (April, 1982) and then to about 3 Hertz (Sept, 1982). This reduction in frequency agrees with that predicted for the frequency of the second mode with changes in the thermal shield support system (Fig. 2).

There were no significant changes in the in phase peaks at 13 Hertz and 19 Hertz (Only the data of September, 1982 are valid below 2 Hertz due to filtering). The out of phase data shows no change in the 7.5 Hertz peak associated with core support barrel motion. These observations correspond to those predicted for the n equal 1, 3 and 4 modal frequencies (Table 2) as thermal shield support conditions are changed.

6.0 SUMMARY and CONCLUSIONS

The objective of this paper was to present results of a post damage analysis of the LPM and IVM data for St. Lucie, Unit 1.
FIGURE 7  POST OUTAGE ANALYSIS OF ST. LUCIE 1 IVM DATA
The LPM data was reanalyzed employing a Modified Amplitude Probability Distribution. This resulted in an improved representation of the changes in the impact related signals with operating time.

The IVM data was reanalyzed using phase separation in conjunction with a modal analysis of the core barrel - thermal shield to simulate changes in modal frequency with assumed changes in support conditions.

The results, when integrated with the remainder of the failure analysis investigation, provided the basis for the postulated sequence of events which may have led to thermal shield damage (Figure 8).
Two general conclusions can be drawn from this study: that advanced analysis techniques are available which can be used to improve the evaluation of IVM and LPM data, and the necessity of performing a modal analysis of the core support barrel system to support the evaluation of IVM data.

REFERENCES


THE USE OF EXCORE NEUTRON NOISE AT NEAR ZERO REACTOR POWER TO MONITOR THERMAL SHIELD SUPPORT SYSTEM INTEGRITY

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Keywords:
Thermal shield, excore detectors, neutron noise, near isothermal conditions, phase separated PSD.

Abstract

Several nuclear reactors which incorporate a thermal shield design have experienced degradation of the thermal shield support structure. Combustion Engineering, Inc. (C-E) and the Omaha Public Power District (OPPD) have developed both equipment and a surveillance program which monitors the performance of the thermal shield support structure at the OPPD Fort Calhoun nuclear station.

Background:

Both the ASME and the American National Standards Institute (ANSI) in their joint standard have accepted a consistent program of Internals Vibration Monitoring (IVM) using neutron noise to be a reliable reactor internals surveillance method (Ref. 1). The U. S. utility, OPPD, has developed an administrative procedure to monitor reactor internals vibration levels using existing excore neutron detector signals at the Fort Calhoun station, located north of Omaha, Nebraska.

The Fort Calhoun reactor (Figure 1) is a 502 MWe Pressurized Water Reactor (PWR) designed by C-E. In operation since late 1973 the reactor internals include a Core Support Barrel (CSB) and a Thermal Shield (TS). During the scheduled ten year In-service Inspection (ISI) performed in 1983 a visual examination was made of the CSB/TS support structures. The examination indicated that all support components were in good condition. OPPD has been actively involved in a regular program of IVM surveillance of reactor internals motion since 1985.

![Figure 1. Ft. Calhoun Reactor](image-url)
Core Barrel Thermal Shield Description:

Figure 2 details the arrangement of the Ft. Calhoun reactor internals. The thermal shield is a right circular cylinder concentric with the core barrel. The thermal shield is attached to the core barrel and extends the length of the active core. The thermal shield is suspended from support lugs on the core support barrel. Radial positioning is provided by two sets of positioning pins. Figure 3 describes the CSB/TS connection scheme.

CSB/TS Dynamic Behavior:

Because of the support lugs and the radial positioning pin the CSB/TS form a closely coupled mechanical structure which exhibits identifiable beam and shell modes of vibration. The beam bending mode (Figure 4a) is a cantilever mode of vibration of the CSB, similar to a simple beam with one end free and one end clamped.

In this mode the CSB cross section remains circular and translates. The shell modes of vibration (Figure 4b, 4c) are vibration modes involving circumferential variation in the shape of the CSB.

Figure 4A.
Beam Mode Vibration of a PWR Core Barrel

Figure 4B.
First Shell Mode (COS 2e).
Vibration of a PWR Core Barrel

Figure 4C.
COS 3e Shell Mode Vibration of a PWR Core Barrel.
TABLE 1
Predictions of In-Water Modal Frequencies (Hertz)

<table>
<thead>
<tr>
<th>Mode</th>
<th>Nominal</th>
<th>All Pins Removed</th>
</tr>
</thead>
<tbody>
<tr>
<td>1. Beam</td>
<td>7</td>
<td>7</td>
</tr>
<tr>
<td>2. COS 20</td>
<td>12.5</td>
<td>7.9</td>
</tr>
<tr>
<td>3. COS 30</td>
<td>16.3</td>
<td>14.9</td>
</tr>
<tr>
<td>4. COS 40</td>
<td>22.8</td>
<td>22</td>
</tr>
</tbody>
</table>

Column #1 in Table 1 lists the calculated in-water modal frequencies of the Fort Calhoun CSB/TS (Ref. 2). These frequencies can be identified in a typical Auto Power Spectral Density (APSD) taken from the Ft. Calhoun reactor at 100% power (Fig. 5).

Analysis of the CSB/TS support structure (Ref. 3) indicates that a loss of radial positioning pin effectiveness can initiate the decoupling of the CSB/TS. This decoupling effects vibration modal frequencies. The detection of radial positioning pin condition is possible using the neutron noise portion of the existing linear power range detectors.

Column #2 of Table 1 lists the calculated in-water modal response frequencies for the Ft. Calhoun CSB/TS in the case of a postulated loss of effectiveness of all radial positioning pins. Based on comparison between "Nominal" and "All pins removed" cases, it can be observed that only the first shell mode of vibration (COS 20) changes significantly from its nominal 12.5 Hz down to 7.9 Hz with all pins removed. By monitoring the frequency of the CSB/TS first shell mode one can infer changes in positioning pin effectiveness.

![POWER SPECTRAL DENSITY](image)

Figure 5. Typical 100% Power Ft. Calhoun APSD

Reactor Power and Positioning Pin Loading:

A number of factors influence the effectiveness of the radial positioning pin including mechanical preload and differential thermal expansion, both of which increase the coupling force between the thermal shield, and the CSB. The pressure differential across the thermal shield, support lug bending and radiation induced relaxation all decrease the CSB/TS coupling force. The thermal expansion force is the only positioning pin loading force which can be controlled during reactor operation. Increased temperature differential between the core support barrel and the thermal shield increases the interference between the positioning pin and the CSB.
Power Level Selection:

In order to monitor positioning pin effectiveness it was desired to select plant operating conditions for which the CSB/TS would be most susceptible to any reduction in positioning pin effectiveness. If the reactor is in an isothermal condition the thermal expansion force is zero and the mechanical preload given the positioning pins during installation would be the sole force maintaining a positive load on the positioning pins.

Zero reactor power produces ideal isothermal conditions between the CSB/TS components. However, zero reactor power does not provide an adequate neutron flux to allow the linear power range excore detectors to be used. If a simple measurement scheme using existing instrumentation is desired then a near zero power level must be selected which:

1) Produces a minimal thermal expansion force.
2) Provides an adequate excore detector flux signal.

The thermal expansion force exerted by the CSB on the positioning pins is:

\[ F_{\text{therm}} = K \times (R_0 + \frac{L}{2}) \times \alpha \times \Delta T \]

- \( K = \) CSB/TS stiffness
- \( R_0 = \) Core barrel outer radius
- \( \alpha = \) Coefficient of thermal expansion for the core
- \( L_0 = \) Positioning pin length
- \( \Delta T = \) Temperature difference between CBS and TS.
- \( \) barrel- thermal shield.

Since \( \Delta T \) is nearly linear with reactor power the thermal expansion force is also nearly linear with reactor power. Using this linear relation the thermal expansion force at 5% reactor power would only be 1/20th the 100% thermal expansion force. From a thermal expansion force consideration the lower the reactor power the better. Reactor power below 5% does not produce significant thermal expansion force.

The other consideration in selecting a reactor power level was would there be sufficient vibration related neutron noise signal to allow a successful measurement using the plant linear power range detector signals. The background noise level is made up of uncorrelated noise sources, not related to the vibration induced fluctuations, found anywhere along the excore detector signal path. This background level has an APSD which is constant at all frequencies and has a constant voltage value. The vibration related fluctuations measured by the excore detectors are linear with reactor power. Our measured signal is their sum of:

\[ X(\text{signal}) = \text{vibration (power)} + \text{BACKGROUND (constant)} \]

Since the background level is constant it becomes a progressively larger percentage of the measured signal as the reactor power level decreases. Figure 6 is a composite of
APSD's for three different power levels (30%, 10% and 5%). In all three traces the vibration portion of the signal remains a relatively constant percentage of the D.C. power. The background noise is constant regardless of reactor power level and therefore consistently becomes a larger percentage of the signal as the power level decreases.

A problem exists with measuring shell mode frequencies. The shell mode vibration scale factor, (% neutron flux/mil of motion), is smaller by a factor of three (3) than the scale factor for the beam mode (Ref. 4). Coupled with this smaller shell mode scale factor is the smaller shell mode vibration amplitudes. Together these two factors produce a detected shell mode neutron flux signal which is about a factor of one hundred (100) smaller than the beam mode response in the APSD. Examining Figure 6 at the beam and shell frequencies it is quite easy to identify the beam mode frequency at all three power levels. However, it becomes increasingly difficult to identify the first shell mode as the reactor power decreases.

Five percent reactor power was chosen to be the plant power level for the positioning pin effectiveness measurement. The five percent power level is both the highest power level at which thermal conditions are not appreciable and the lowest reactor power at which CSB/TS shell mode frequencies can be detected.

Data Acquisition System:

The measurement of reactor internals vibration data requires specialized signal conditioning equipment. C-E has developed an Internals Vibration Monitoring (IVM) system which provides all of the necessary electronic equipment and analysis functions needed to perform regular IVM measurements. The IVM system contains a Signal Conditioning Module (SCM) which houses the processing electronics, a portable computer, which controls the SCM and computes the analysis functions, and a graphics hard copy printer.

The Signal Conditioning Module accepts two channels of ex-core signals as inputs and outputs the bandlimited and amplified vibration fluctuations which make up less than 0.25% of the gross power level ex-core signal.

The C-E SCM (Fig. 7) provides:

- High pass filtering for elimination of the gross DC power level.
- High gain PRE-filtering amplification
- Low pass filtering for frequency limiting the data
- High gain POST-filter amplification
- Controlled sample rate Analog - to - Digital Conversion (ADC).

The SCM is totally controlled in all its ranges and functions by the computer.

The output of the SCM is then processed by the IVM analysis software package and analysis results are stored on both disk and hardcopy graphic printer output.

![Diagram of C-E Internals Vibration Monitoring System Signal Conditioning Module Block Diagram](image-url)

**Figure 7.** C-E Internals Vibration Monitoring System Signal Conditioning Module Block Diagram
The C-E IVM analysis software acquires both time and frequency domain data. The standard IVM analysis functions include: Auto and Cross Power Spectral Density functions, signal Coherence and Relative Phase functions as well as user specified band select RMS (root-mean-square) energy calculations. Also available is the implementation of spectral phase separation algorithms. The IVM software can also recall previous computed analysis results to allow comparison between two analysis data sets.

Near Zero Power Field Measurement

In June 1986 a near zero neutron noise measurement was made at Ft. Calhoun plant. This measurement was made at nominally 5% reactor power using the C-E IVM system. Two linear excore detectors 180° apart were measured and analyzed. The signal obtained from the first shell mode exhibits an in-phase relationship between detectors which are 180° apart (Fig. 8). In addition to the normal spectral analysis a special phase separated APSPD (Ref. 5) was calculated. This phase separation technique applies to the APSPD processing for which the measured APSPD is the sum of two processes which are either in phase (0°) or out-of-phase (180°) relative to each other.

For this special case the measured APSPD can be separated based on phase into APSPD (0°) and APSPD (180°). Since our interest was in the CSB/TS first shell mode frequency, the in-phase, APSPD(0°) spectra, would be the most sensitive to change of that vibration mode. Figure 9 is the 0°-phase APSPD from the 5% reactor power measurement at Ft. Calhoun. The frequency of this vibrational mode is still noted to be in the same 10-12 Hertz region as in the 100% power APSPD (Fig. 5). The inference from this analysis was that the CSB/TS support system was effective.

Figure 8.  Excore Detector Phase Relationships for First Shell Mode CSB Vibration

Figure 9.  Ft. Calhoun 0°-Phase APSPD at 5% Reactor Power
Inspection Deferred Through Monitoring

In 1984, the U. S. Nuclear Regulatory Commission (USNRC) concern over CSB/TS support system problems led OPPD to commit to the USNRC to perform an examination of the Ft. Calhoun CSB/TS during the 1987 maintenance outage. This examination would have been performed four years after the normal ISI ten year inspection. OPPD petitioned the USNRC in August 1986 for deferral of this examination until the next scheduled ISI inspection to be performed in 1993.

The deferral effort consisted of both an analytical effort and a commitment to an IVM surveillance program. Emphasis in the deferral effort was also given to the positive results of the near zero power IVM measurement taken in June 1986.

In February 1987 the USNRC granted OPPD a deferral from CSB/TS examination until the scheduled 1993 ISI date. OPPD's commitment to the USNRC included the analysis of IVM data on a quarterly basis and collecting near isothermal IVM data once per fuel cycle.

Conclusion

The early detection of thermal shield support system structural change is the primary goal of the OPPD IVM program at Ft. Calhoun. IVM measurements at near zero reactor power provide an early warning of a possible change in radial positioning pin effectiveness by monitoring changes in the CSB/TS vibration characteristics. The OPPD surveillance program provides a reliable early warning of potential structural problems with a minimal impact on plant operations.

References

SYSTEMS (PART I)

Session chairman: K. Dach (C.S.S.R.)
SUMMARY OF THE SESSION

The common feature of all reports presented in this session was the tendency to automatize the surveillance procedure to eliminate the necessity of human expert presence during results evaluation and to introduce step by step artificial intelligence into systems.

A microprocessor-based system KWO presented by Jax and Juthrof was shown to be very effective for detecting, classifying and evaluating acoustic signals. The knowledge base of the system is on the high-level of actual know-how of experts so that external experts could concentrate their efforts only on the few cases of anomalous noises.

Valko discussed in depth the problems connected with integrating noise analysis into reactor monitoring systems. The integrated noise analysis system is supposed to combine two basically different approaches to noise measurement interpretation. The pattern recognition method, which is good at detecting occurrences, gives little information about the physical or technical nature of the phenomena and the analytical method, using physical models reproducing the measured noise parameters. A very important part of this system is signal validation procedure and its incorporation into automatic operations of the system.

van Niekerk and Sunder introduced an on-line condition monitoring system (COMOS) which has been developed for main coolant pump shaft vibration monitoring as well as for general vibration monitoring of passive primary components. A high degree of automation and information extraction and a user-friendly representation of measurement results allow utility people to apply early failure detection methods without continuous expert assistance. The modular hardware and software makes possible the future integration of noise rules to increase the diagnostic possibilities of the system.

The problem how to utilize large masses of data stored on hard disk to give very specific types of information in answer to narrow and precisely formulated questions. It was recognized that computing efficiencies would be achieved by storing spectra features rather than numerous spectral points in the databases. For using the described approach the following comments are proposed: User-friendly programs with numerous and powerful commands should be used. One should be aware of the possibility of more information existing in the individual data points and should not always restrict analysis to just a few features. Close contact with the data by visual examination of samples should be maintained.

The paper of March-Leuba and King described the characteristics of a portable real-time system used for nonperturbational measurements of stability in boiling water reactors. The algorithm used in this system estimates the closed-loop asymptotic decay ratio using only the naturally occurring neutron noise and it is based on the univariate autoregressive methodology. Experimental evidence has shown that open-loop decay ratios can be as much as 50% smaller than closed-loop decay ratios and than they are nonconservative estimates of the reactor's stability.
ADVANCED TECHNIQUES FOR THE SURVEILLANCE OF LIGHT WATER REACTORS USING MICROPROCESSOR BASED SYSTEMS

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Abstract - Monitoring systems are based on complex and high-technology methods and - in the past - needed the full skill of well-trained engineers. A new generation of microprocessor based systems is presented which improve the reliability and quality of the techniques and relieve the efforts and number of the staff needed. The systems are stand-alone systems with automatic routines of calibration, measurement and defined evaluation procedures, with an improved data storage of the relevant measuring parameters and with special analysis software, which can be used by interactive access of the experts. An overview of the monitoring systems of KWU is given and the principal features are explained on two examples.

1. OBJECTIVES
Monitoring Systems are part of an overall concept of assuring the integrity of the primary coolant systems of nuclear power plants.

The objectives lie in both safety-related and economic considerations:

- Early detection of faults and hence minimization of damage, facilitation of fault clearing
- Prevention of sequential damage
- Reduction of inspection costs and radiation exposure.

Monitoring systems are considered to be one link in the chain of provisions for ensuring component integrity and high availability of KWU reactors. They are to be classified as information systems e.g. their signals do not cause any automatic reaction of the reactor protection system, but may naturally lead to a decision to shut down the reactor, if a serious situation has been evaluated. There is a very small probability of the indicated faults (leaks, loose parts in the primary circuit, loosening of bolts inside the reactor pressure vessels etc.) and in many cases the faults are detected in a very early stage, so that suited measures can be postponed to a later stage (e.g. to the annual revision of the plant) or be done in a more economic way.

A survey of the systems concerned, their function, the measured variables to be acquired, evaluation parameters and the status of development is given (Fig. 1).
<table>
<thead>
<tr>
<th>Function</th>
<th>LPMS</th>
<th>VMS</th>
<th>LDS</th>
<th>LMS</th>
<th>FAMOS</th>
</tr>
</thead>
<tbody>
<tr>
<td>Detection and identification of loose parts</td>
<td>Detection of changes in mechanical integrity such as loosening of bolts, fracture of springs, vibration of piping</td>
<td>Detection and monitoring of leakages in the reactor building</td>
<td>Detection, localization and sizing of leaks in specific pressurized systems</td>
<td>Detection of transient thermal stress determination of the fatigue usage factor</td>
<td></td>
</tr>
<tr>
<td>Approximate localisation of component parts</td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
</tr>
</tbody>
</table>

<table>
<thead>
<tr>
<th>Primary dynamic measured variables</th>
<th>Structure burst signal</th>
<th>Mechanical vibrations</th>
<th>Humidity in the compartments</th>
<th>a) High frequency leakage noise in structure</th>
<th>Temperature at the outside surface, respectively process instrumentation</th>
</tr>
</thead>
<tbody>
<tr>
<td></td>
<td></td>
<td>Pressure fluctuations</td>
<td>sump level</td>
<td>b) Moisture level</td>
<td></td>
</tr>
<tr>
<td></td>
<td></td>
<td>Neutron flux noise</td>
<td>airborne activity</td>
<td></td>
<td></td>
</tr>
</tbody>
</table>

| Frequency                                                               | 1–10 kHz                 | 0.1–200 Hz               | 0.100–600 kHz               | < 0.25 Hz                                    |                                                               |

| Evaluation parameter                                                   | Parameters of burst signals such as rise time, amplitude, etc. | Frequency spectrum, coherency function | Increase of humidity, sump level or activity versus time and position, etc. | a) RMS amplitude as function of time and measuring point | a) Local monitoring temperature field at the outside surface precomputed stress fields |
|                                                                        |                          |                          |                             | b) Moisture level as function of the pumping time (position) | b) Global monitoring service condition and specified stress characteristics |
|                                                                        |                          |                          |                             |                                               |                                                               |
|                                                                        |                          |                          |                             |                                               |                                                               |
|                                                                        |                          |                          |                             |                                               |                                                               |

| Status of development                                                  | Microprocessor system 1985 | As LPMS                   | Analog system 1981           | a) Prototyp 1982                           | Prototype                     |
|                                                                        |                          |                          |                             | b) Prototyp 1985                           |                                                               |

---

**Overview of the Monitoring Systems**

**Fig. 1**

2. **SYSTEM CONCEPT**

The monitoring systems are based on a common underlying principle:

- Dynamic measured variables are continuously or periodically acquired and conditioned appropriately.
- The conditioned data are compared with predetermined characteristics or limits.
- A significant deviation indicates an anomaly in the plant or system condition.
The common technical concept, which is shown in Fig. 2, provides for autonomous modular individual systems which, however, are capable of transferring their data to a host computer which may be designed as a stationary or mobile system.

The salient features of such a system concept are enumerated below:
- Uniform electronics
- Continuous mode of operation
- Automatic calibration
- Control, monitoring and documentation by microprocessor
- Detailed documentation of incidents inclusive of preincident history
- Inclusion of operating data, operating mode identification and as necessary, of alarm suppression

Requirements relating to vibration monitoring, loose-parts and leakage monitoring are formulated in general terms in the RSK-Guidelines and KTA-Safety Standards (3201.4, Section 8). On account of these requirements, systems and facilities for loose-parts, vibration and leakage monitoring are provided in the majority of German nuclear power plants.

The many years of experience mostly based on analog techniques, with these systems, formed the basis for further development and led finally to the above system design.
As described in some detail for the loose-part and acoustic leak monitoring System (LPMS and ALMS) in this paper and the other monitoring system in 1, 2/ the introduction of modern digital techniques - together with adapted sensors and calibration procedures - improved the capability and handling of the systems. By means of digital preprocessing and microprocessors more relevant data (more signal and new signal parameters respectively) are collected and special intelligence on signal conditioning and evaluation procedures are implemented. The digital control of measurement and evaluation reduces manpower for systems' operation. In most cases a physical calibration routine including the sensor is implemented leading to a higher quality control of the system during operation.

3. LOOSE PART MONITORING SYSTEM (LPMS)

The task of the LPMS is to detect loose parts entrained in or at least set in motion by the coolant. The system picks up structure-borne acoustic signals in the audible range (< 20 kHz), which originate upon impact of parts on walls or internals using piezoelectric accelerometers. Monitoring encompasses the primary system of LWR; typically 14 and 8 sensors for the PWR and BWR respectively (1300 MW) are installed.

3.1. System description

The signal of the sensors pass through wideband preamplifiers and isolation amplifiers (preamplifier station in the compartment) to the main system outside of the containment (Fig. 3).

---

**Fig. 3**

**Hardware**
There, a bandpass filter limits the frequency range of the signal for further internal processing (1-10 kHz) in order to increase the signal/noise-ratio. At this point the signals are digitized and led to a multichannel transient recorder. System triggering and alarm output are initiated once threshold values of the averaged RMS-signal (integration time of 1 sec) are exceeded at one or more channels whereas fixed and variable limits are implemented. In this case the transient recorder hands over the time signals of all (up to 16) channels to the microprocessor (including digital tape, printer etc). For the purpose of diagnosis, the signals are recorded with a high time resolution and a pre-trigger-time and the burst patterns thus obtained are evaluated. The information needed for localisation and interpretation of the results is generally obtained with the aid of signal parameters, transit time differences and statistical functions. The corresponding software (Fig. 4) is organized as follows.

**KÜS/LPMS Event Processing and Documentation**

Firstly, an alarm protocol and alarm diagramm of each event is automatically stored and printed showing the individual time signals with a standardized resolution and the main measuring parameters such as date, time, trigger channels, type of triggering, total numbers of alarms and peak amplitudes per channel, level of background noise, amplification etc. Thus, most of the information needed for a quick-look and first evaluation is given. Secondly, special evaluation software can be activated by the authorized user.

- for a more precise analysis of the individual signal parameters (leading edge decay-time) and the transit time differences at various channels by using zooming and cursor functions,
- for evaluations statistics such as burst rate histograms (number of events per time interval vs time), amplitude distributions (Fig. 5) and distributions of time differences and
- for listings of the recorded events (amplitude, trigger channels, date and time of occurrence etc.).

Running the evaluation software, new incoming signals are still recorded and documented as data recording has always a higher priority within the software routines.
In case of a high event rate
- the total number of all events is recorded by an additional alarm modul in the transient recorder at any rate
- the transient recorder might be switched to the storage of the short-time RMS-signal (integration time of ca. 5 ms) representing a proper envelope of the original signal (buffer of further 8 events).

Calibration procedures can be easily handled by the system (via a remote control unit RCU) by
- either transmitting an electrical signal (wide noise or sinus of variable amplitude) to the input of the preamplifier
- or activating a new designed impact hammer which produces a defined acoustic signal to the structure and allows the control of the sensors at any time, too.

In principal the hammer is a closed chamber with an internal loose part (130 g), which hits the bottom by a defined velocity (6 m/s). The contact time of this impact can be measured as it is a control parameter of the physical strength of the impact.
The hammer is designed and tested for elevated temperatures in the environment of LWR and for a high number of reproducible impacts (> 10 000).

3.2. Experience
In September 1987 five microprocessor based systems has been installed in nuclear power plants and five further systems are under construction. The experience, gained till now, can be summarized as follows. The new system showed to be very effective for detecting, classifying and evaluating acoustic signals.
Much more signals and data were recorded than this was possible by the older analog systems. Therefore more information was gained on acoustic burst signals caused by operational conditions (frictional noises by thermal expansion in the starting period of operation, cavitation etc), which has to be distinguished from impact signals of loose parts.

In one case small signals were detected indicating loose parts. From the burst pattern, the triggering channels, the statistics and listings it become clear that
- there is an anomalous noise, which has to be analysed as impacts
- the sources are coming from the upper region of the RPV but are distributed along the circumference to a few positions
- the amplitude and burst rate was not very strong.

No immediate reaction to the operation of the plant was necessary, nevertheless further analysis and closer observation on this signals were ordered, which are still going on.
In this and other examples the system proved to be easy to operate giving a quick and precise information on the intensity and characteristics of the signal avoiding false alarms. Therefore, external experts of LPMS could concentrate their efforts only on the few cases of anomalous noises.
Calibration showed to be rather easy and efficient using the new impact hammers. Installing one hammer for each loop (PWR) on the pipe section between RPV and steam generator the peak amplitude of the calibration signals could be adjusted up to 2 to 8 g on all sensors in the primary circuit (Fig. 6). This level allows it to check the complete system including the sensitivity of the sensors even under operational conditions of the plant. Furthermore, the exact time differences of the signals at various channels and thus, the sound velocities can be easily measured by the cursor function of the system.

3.3. Outlook
For purpose of diagnosis and efficiency, further analysis software is planned to be implemented:

- determination of frequency spectra for characterizing the background noises and impact signals
- measurement of time differences by cross correlation techniques
- implementation of planar and linear source location algorithms (known from acoustic emission tests) for the RPV and the other components respectively.
- determination of the distance between source and sensor by detailed mode analysis and measurement of the time differences between the filtered components of one signal /4/.

The techniques has mostly been proved in case by case on real impact signals and will mainly help to improve location accuracy. The last, rather complex method (together with a planar location on the surface) enables a more precise location on impacts at internal constructions of the RPV or steam generator.

4. ACOUSTIC LEAK MONITORING SYSTEM (ALMS)
In addition to the conventional leakage monitoring facilities (for radioactivity, ambient humidity condensation, etc.) an acoustic leakage monitoring system ALMS has been developed and prooftested as prototype in an German nuclear power plant. The acoustic leakage monitoring system is based on the principle that escaping fluid induces high-frequency structure-borne noise of a continous signal characteristic in components and that this noise is detected by externally fitted probes /3/.

The advantages of this system lie in:

- the promptness of the leakage indication (< 2 min)
- adequate localization of the leakage (≤ 1 m) by comparision of the leakage noise (rms value) at various measurement locations (Fig. 7)
- capability of detecting both external and internal leakages (therefore also suitable for valve monitoring)
- high detection sensitivity (100 kg/h at the primary circuit of a PWR, corresponding to a leak size of ≈ 1 mm²).

The system is based on the following design concept (Fig. 8):

- use of piezoelectric emission probes with waveguides which are simple to backfit
- distribution of sensors as appropriate to background and local conditions
- preamplification of the signals
- determination and digitizing of the r.m.s. value of the signals (in substations) and transmitting of this data to a central microprocessor
- comparision of the noise level with adjustable threshold levels (fixed and floating thresholds)
- if these are exceeded alarm annunciation
- determination of leak noises (proper substraction of background noise)
- printout of noise history (including a relevant time interval before leak alarm) and location diagram of leak noise (Fig. 7).
KÜS – Peakamplitudenverteilung

Beginn: 01/01/87 00:00:00  
Ende: 29/01/87 00:00:00  
Trigger-Kanäle: 1 2 3 4  
Modus: Alarm  
Signaltyp: Original

Nur vom Rechner bestimmte Maxima  
Zeit + Mode + Type o.k. 44  
Triggerkanal o.k. 22  
max. Amplitude in mG 30000

KÜS: Peak Amplitude Distribution

Fig.5

KÜS/LPMS

Event 2079 02=VRAR170  
A. 59 mG  
-10 -5 0 5 10 15 20 25 30 nsec

Event 2079 04=VRAR8  
A. 59 mG  
-10 -5 0 5 10 15 20 25 30 nsec

Event 2079 05=VRAR120  
A. 44 mG  
-10 -5 0 5 10 15 20 25 30 nsec

Event 2079 06=VRAR240  
A. 44 mG  
-10 -5 0 5 10 15 20 25 30 nsec

Event 2079 08=EV2 AR  
A. 183 mG  
-10 -5 0 5 10 15 20 25 30 nsec

Event 2079 11=EV2 AR  
A. 6 mG  
-10 -5 0 5 10 15 20 25 30 nsec

Fig.6
From the microprocessor a noise generator, to which transmitter probes are connected, is periodically switched on to simulate an acoustic leakage noise in the structure and test both the calibration of the measuring chains and the proper functioning of the system. The calibration routine includes a mathematical procedure by regression to determine the attenuation factor of leak noise and a correction factor for each measuring channel considering individual sensor sensitivity. It is normally actuated every day and can be documented in detail, especially if relevant changes of the factors are determined.

ALMS records any change in the noise level of the plant in the frequency range of 100 - 400 kHz and therefore supplies information on the plant condition beyond the detection of leakages (Small loose parts, change of flow condition etc.). The method has been excepted as an alternative leak detection system in Germany. Today two modern microprocessor based systems are under construction and two further systems are planned for power plants abroad.

5. SUMMARY AND PROSPECTS

Further and new developments of the systems described allows for the many years of experience in the fields of loose parts, vibration and acoustic leak monitoring.

The foreground of hardware development is occupied by the modular concept, earliest digitizing of signals as necessary and the use of modern microprocessors with their manifold capabilities. This includes especially the tool to implement special, but complex Know-How of experts into automatic higher-level analysis routines of the system - a procedure of development, which has been started, but should be updated from time to time. Other objectives of development lie in the multiple application or integration of sensors in different systems and the combination of information of different systems.
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INTEGRATING NOISE ANALYSIS INTO REACTOR MONITORING SYSTEMS

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Abstract - Three operating PWR units in Hungary have been equipped with reactor noise measuring systems, installation at the fourth unit has been completed. The experience with the existing systems is encouraging, attention is now focused on the development of an evaluation/interpretation system that is connected to the reactor monitoring computer. Noise signal selection and identification will be program controlled, reactor status information will be obtained from the process computer. Evaluated noise reports will appear on the operator's displays. Continuous monitoring will be governed by default sequences, interactivity will be provided at operator level and plant physicist level. An expert system will combine formal (statistical) and analytical (physical model based) interpretation methods.

1. INTRODUCTION

Power reactors operating almost continuously at nominal power provide only very limited access to dynamic information with the exception of the possibility of measuring and analysing the fluctuation (noise) inherently present in practically all stationary process signals. The diagnostic value of the information contained in the noise has long been acknowledged and led to the installation and use of some kind of noise measuring equipment at almost all nuclear reactors. PAKS Nuclear Power Station in Hungary being no exception. This paper gives a brief account of the experience obtained through the application of noise analysis at this 4 reactor plant. It is shown that a more sophisticated system that is integrated into the monitoring or information system of the reactor may increase the power of noise diagnostics, while performing and using the noise measurements become easier.

The noise analysis programme accomplished so far consisted of several phases. It began with the installation of vibration, pressure, neutron flux and temperature measuring devices and with theoretical investigations to support the interpretation of the measurements, it was followed by regular noise recordings and off-line analysis. An extensive set of baseline data has been accumulated, the most significant features of the spectral descriptors have been identified and interpreted in terms of reactor and primary loop component and process parameters. The capability of the methods for detecting and identifying anomalous conditions has been demonstrated.

Regular and efficient use of the noise information now requires highly skilled personnel because measurements and recordings with add-on devices are complicated to use and the results in the form of power spectra, phases, coherences, etc. are difficult to understand. The development of evaluation and interpretation methods coupled with a more user-friendly hardware is planned to form a next generation of the noise diagnostics system that will perform a series of functions automatically and the noise information will be interpreted, together with standard stationary plant information, and the results presented to the operator in an easily comprehensible form.
2. NOISE MEASURING EQUIPMENT

Paks Nuclear Power Station reactors and primary loop components are relatively well instrumented for noise measurements. The signals available for recording or analysis at the different Units are shown in Table 1.

<table>
<thead>
<tr>
<th>type of signal</th>
<th>number of signals at Unit</th>
</tr>
</thead>
<tbody>
<tr>
<td></td>
<td>1</td>
</tr>
<tr>
<td>in-core neutron</td>
<td>1x7</td>
</tr>
<tr>
<td>ex-core neutron (IK)</td>
<td>2</td>
</tr>
<tr>
<td>ex-core neutron (JK)</td>
<td>3</td>
</tr>
<tr>
<td>temperature</td>
<td>-</td>
</tr>
<tr>
<td>low freq. pressure</td>
<td>3</td>
</tr>
<tr>
<td>pressure</td>
<td>9</td>
</tr>
<tr>
<td>acceleration</td>
<td>42</td>
</tr>
</tbody>
</table>

* up to 3x7 detectors can be switched simultaneously to the noise analysis system

Table 1.

The accelerometers and the pressure transducers have been installed solely for noise diagnostic purposes. The other sensors are part of the reactor standard instrumentation. This fact makes their noise analysis application more problematic. Because the ionization chamber's primary function is in the safety system, a method had to be developed for sharing the signal. The standard handling of ionization chamber currents at this plant is based on current to frequency conversion, for noise measurements a reconversion of the varying frequency pulse train to analog voltage is applied. The analog signal obtained has been examined and proved to be well suited for noise analysis. Instead of sharing of SPND currents, a relay circuit was interconnected to switch the signals alternatively to the in-core flux mapping of the core monitoring system or to the noise analysis equipment. The switching is initiated manually, because noise measurements are only permitted at specified intervals. The thermocouples which are located in the reactor vessel at the exit of the fuel bundles are also part of the core monitoring system, their signals, however, could be connected simultaneously to both systems by suitable electronics.

Noise signals are particularly sensitive to inadequate signal conditioning and improper signal paths, therefore systematic studies were needed to determine the best locations for preamplifiers. In Units 1 and 2, the SPND noise preamplifiers were put close to the detectors in a non-servicable place, but in Units 3 and 4 they are located further away, since the investigations showed that signal quality allowed this. The preamplifiers for the pressure and acceleration detectors are located, in all Units, close to the detectors in non-servicable positions.

For detailed vibration monitoring in VVER-440 reactors, because of the 6 loops and large number of primary loop components, many detectors are needed. It can be problematic to locate the detectors and find free cables out of the containment if the use of these signals is decided at a later stage of planning or only during the construction of the plant.
Integrating noise analysis

Two noise laboratories were set up, where the noise signals can be accessed for measuring or recording, one for Units 1 and 2, one for Units 3 and 4. In these laboratories there are all the main amplifiers, filters, RMS meters, visual inspection screens, etc., and spectrum analysers or analog tape recorders can be connected to the signals. On-line analysis, up to now, has been limited to simple measurements like spectrum comparisons and qualitative observations. Systematic evaluation of the noise signals were accomplished by multichannel analog recording and subsequent off-line analysis.

3. THEORETICAL AND METHODOLOGICAL INVESTIGATIONS

By noise measurements one rarely can measure directly the parameters of interest, it is more likely that to sort out the answers to questions of diagnostic significance from the abundance of information that is contained in the noise measurements, sophisticated evaluation and interpretation procedures are required. From this point of view, pure mechanical vibration measurements are easier to follow, qualitatively, at least. For a quantitative interpretation, models of the coupled mechanical structures are needed. Neutron noise can "see" a great variety of parameter changes at large distances, but to identify the neutronic-thermal-hydraulic reactor and primary loop models are necessary. It is theoretical models to the stronger, dominating at least in given ranges of frequency.

Following the above, general considerations, a programme of theoretical model development and model investigations have been carried out, supported from time to time by dedicated zero power experiments, to acquire a general understanding of the noise sources and transfer functions active in a nuclear reactor.

A modified theory of propagating in-core temperature fluctuations was proposed in which the inlet temperature fluctuations and velocity fluctuations were taken into account. The resulting measurable neutron flux fluctuations were interpreted with the assumption of a fluctuating amount of subcooled boiling (T. Katona et al., 1982). Control rods in VVER-440 reactors are in fact control assemblies, fuel assembly size absorbers with fuel followers. Flow induced vibration of these long structures can cause problems, therefore the theoretical study of the vibration and its detection by neutron noise is important. Pazsit and Glockler (1983, 1984) examined the statistical properties of such vibrations and developed a method for the localization of absorber vibration using in-core neutron detector noise signals. Pazsit et al. (1984) found characteristic differences in the displacement probability distribution of vibrating structures according to the nature of the excitation, the theory directly suggests that neutron noise measurements can identify the vibrating absorber and the probable reason of the vibration can be inferred, too. The neutron noise and temperature noise are strongly coupled in the reactor core, theoretical modelling is done by coupled neutron kinetic and thermohydraulic models, as in the work of Mesko and Kozma (1984) and Kozma and Mesko (1985). Aguilar and Por (1987) evaluate the temperature coefficient of reactivity from noise measurements. In zero power experiments, the validity of theoretical models can be checked, under conditions usually far from power reactor conditions. Pazsit and Lux (1982) deal with the problem of how to narrow this gap.

4. EXPERIENCE WITH NOISE ANALYSIS

Earlier publications (Valko et al., 1985, Glockler et al., 1986a, Glockler et al., 1986b) show examples of the various noise measurements taken regularly at PAKS Nuclear Power Station with the existing noise measuring equipment. The results of baseline measurements give the characteristic noise signatures and spectral functions which are very similar at all Units, variations with core life (boron concentration or burn up) are generally understood.

The in-core neutron signals give spectra that are less structured than similar spectra measured at some other reactors. This may be due to the Rhodium emitter of the SPNDs or some other instrumental effect. On the other hand, these signals seem to be quite sensitive to conditions in the core. Vertically placed detector pairs show linear phase, characteristic of coolant propagation, the inferred velocity being close to the design values. The neutron noise is affected
by density changes in the coolant, the coupling becoming stronger when there are steam bubbles present as a result of slight subcooled boiling, therefore, higher coherence and more emphasized lines of phase is expected in the high power fuel assemblies, which is indeed the case. The phase behaviour of the neutron to temperature signal pairs can only be explained by the large time constant of the thermocouples. The in-core neutron detectors located on the same elevation in various fuel assemblies are sensitive to horizontal motion between them, this effect was utilized when an excessively vibrating control assembly was detected and localized.

Ex-core neutron detectors produce signals coherent with the in-core SPNDs at low frequencies, indicating reactivity coupled effects. At somewhat higher frequencies there is an ambiguous indication of core barrel motion. Core barrel motion as measured by the well established method of ex-core neutron detectors has not been reported from VVER-440 reactors, probably because of the mechanical construction of the reactor components being less likely to allow such motion. Our experience shows, nevertheless that core barrel motion would be detected if it became excessive.

The low frequency pressure spectrum contains peaks related to the pressurizer and to the main eigenfrequencies of the loops. This has been established by comparing spectra taken at different temperatures and different water levels in the pressurizer.

There is very little correlation between accelerometer signals and neutron signals, the vibrating control assembly detected by the in-core neutron signals made no visible effect on the vibration signals of the accelerometers located on the guide tubes of the control assembly drives. The frequency of this vibration was about 1.1 Hz, this is quite low for the accelerometers. In addition, the accelerometers pick up a large amount of vibration at various frequencies, only some of them can be identified with known plant parameters. Characteristic pump frequencies are always present, this was identified, among other things, when measurements were taken during the cast down of switched off pumps.

Hammering tests were conducted to investigate the propagation of body sound in the vessel and primary piping to evaluate the chances of detecting loose parts by the installed set of accelerometers. The bursts were quite marked and their successive appearance in the various detector signals show that by proper evaluation techniques such monitoring is possible. For a loose part monitoring that properly covers the whole primary system, however, additional detectors seem to be necessary, the total number of detectors becoming quite large due to the complexity of the primary system.

High frequency acoustic emission detectors are mounted on the vessel head and some other locations and are connected to a separate system of leak detection. This technique is quite promising since coolant leakages are expected to produce high frequency noise which can be distinguished from other noise sources. Inactive tests are still needed to adjust filtering and signal discrimination parameters as well as to verify localization algorithms.

On the whole, five years of experience at PAKS Nuclear Power Station with a growing complexity of noise measuring systems shows the diagnostic potentials of noise analysis, but the need for substantial further development is also apparent. While methods and algorithms are needed for monitoring or measuring certain, clearly anticipated malfunctions like core barrel motion or loose parts appearing in the system, and the number of these “typical” cases still increasing with experience, it is desirable that noise diagnostics be able to detect malfunctions or anomalous conditions of less known origin and when they occur for the first time. Analysis procedures that can cope with this become quite complex and require skilled people, if automatic equipment is used it has to take care of data collection and validation, standard evaluation, archivization of data involving large databases, it has to be able to handle algorithmic and non-algorithmic information with decision making capabilities, etc. One possible approach could be using separate, dedicated equipment for each purpose, another approach, to be discussed in the following, is an integrated system.
5. INTEGRATED NOISE ANALYSIS CONCEPT

The signals that can be made available for noise analysis are largely determined by technical and economical factors. Once the sensors are mounted, it is up to the measuring and analysing system to make the maximum use of the accessible information. In the integrated system, all signals can be connected to the measuring units by program control. This is achieved by analog multiplexing all signals to a so called standardized analog signal board. The analog signals are accompanied by identification and setting parameters in the form of discrete (digital) signals. Selecting and sampling of the analog signals, after the appropriate setting of filters, optional inclusion of additional units like demodulators, etc. are controlled by a lower level processor in the system. Programs that serve this purpose are stored in this level, but the parameters which control this operation are down-loaded from the higher level. The higher level computer runs a multitask hierarchical program system with default sequences.

The basic sequence takes care of the periodic measurement of all signals in given combinations with stored sampling times. Simple spectral parameters are computed, comparisons with baseline data are performed. For each set of measurements stored criteria are used to decide whether conditions are normal or abnormal. According to previously defined tables certain results are retained, these form the database for pattern recognition algorithms. Given combinations of signals are examined for specific information, like the presence of bursts in accelerometer signals indicating loose parts, antiphase in neutronic signals indicating vibration, etc. In the normal, non interactive mode of operation all tests and analyses are performed with a given periodicity and the results are summarized in log files. The control of this sequence is by default data which can, of course, be modified.

The significance of any noise measurement can only be evaluated if reactor and primary loop conditions pertinent at the time of the measurement are known. For this reason a two way communication between the process computer and the noise analysis system is necessary. A table of current reactor parameters is transferred to the noise system and it is stored together with any noise data which are put into archives. It is used by programs which follow trends in the noise descriptors and by all interpretation routines. Summary reports of noise analysis are sent to the process computer to be shown on the operators displays. The operators are not likely to go into the details of spectral descriptors, noise ratios, etc. The information appearing on their display should be as simple as possible. When the conditions are normal, no anomalies were found by the routine noise analysis, this fact is briefly stated. When anomalies occur, or abnormal conditions are suspected, the operators display should offer interactive services with the indication of the suspected malfunction, with fault tree analysis, comparative values taken from earlier measurements and/or taken at other measuring locations. Intelligent color graphics can enhance the power of such systems. Showing uninterpreted spectra to the operator should be avoided, but it is equally unlikely that algorithms could automatically decide whether anomalies are serious or not. The operators should do the decisions but the system must give all the information that is needed, and in the most appropriate form.

The interactive features of the system are planned in order to solve this delicate problem. The automatic monitoring functions ensure that abnormal conditions are observed, but the detailed and systematic search needed for a full explanation of the abnormality is best done by people. And in this work the noise information must be combined with all other plant information. In case of anomalies it is common practice to bring more people, noise analysts among them to the control room. In addition to the operators display, noise experts terminals should be provided which give full excess to the information and full real time control over subsequent data collection and analysis.

The integrated noise analysis system is supposed to combine two basically different approaches to noise measurements interpretation. Algorithms that use a learning period to establish the most relevant statistical properties of signals or groups of signals, and then compare current measurements with the earlier experience, often called pattern recognition methods, can be made rather sensitive to changes yet false alarm rate can be minimized. These methods are good at detecting occurrences but give little about the physical or technical nature of the
problem and specifically about the reasons of the anomalies. This approach can be called formal or statistical as opposed to the other approach which is termed analytical. In the analytical approach one tries to use physical models which reproduce the measured noise parameters. Typical examples are the coupled pendulum vibration models or the core neutronic-thermohydraulic models, which are based on physical considerations therefore, if discrepancies occur, like peaks shifting, etc., the explanation is usually suggested by the model. Different types of sensors gather information from regions of different sizes, this region is always enlarged by working models which account for transfer functions, correlations.

Signal validation is an important task for which the integrated noise analysis system is well suited. Program controlled connection to a large number of different signals is required for this purpose. If signal validation methods are to become part of the automatic operations, it is recommended that as many signals as possible could be connected temporarily to the system, and the noise information and the static information from the same detector and from groups of detectors should be used simultaneously. Abnormal conditions or accidents occurring at the reactor may damage some of the detectors, then the need for signal validation is even more emphasized. In post accident noise analysis the first thing to do is to determine, as much as possible, if anomalous signals come from properly functioning detectors, indicating unusual conditions or they come from damaged detectors. The models used for interpretation can be generalized to cover a wide range of abnormalities, correlation flow meters are likely to give information on the occurrence and location of primary loop leakages in the case of smaller breaks, pressure fluctuations are sensitive to the gas content in pipes and vessels, etc.

An integrated reactor monitoring and noise analysis system communicate with the operator through technology mnemoschemes, graphs, tables and other graphical representations with up-to-date measured and calculated values, and in terms of the most essential physical and technical processes, even if they are complicated and not directly measurable. Instead of using local thumb-rules about the usual and unusual indications of simple panel instruments, the operators continue to think in terms of the physics of the processes with parameter values representative of whole systems rather than single measurements. In this environment the noise parameters can become as familiar as any static indication. The contribution of noise analysis to reactor safety is not limited to statements of conditions being normal or abnormal. It widens the range of physical information continuously available to the operator, it brings much closer the processes and equipment, it helps to understand better what is going on, therefore it helps to follow when anything goes wrong.

REFERENCES


COMOS—AN ONLINE SYSTEM FOR PROBLEM-ORIENTATED VIBRATION MONITORING

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ABSTRACT

The monitoring of the time dependance of the vibrational behaviour of mechanical components is essential for malfunction detection. In contrast to passive primary components of PWRs like pressure vessels and steam generators, mechanical failures in rotating equipment may develop rapidly. An on-line Condition MOonitoring System (COMOS) has been developed for main coolant pump shaft vibration monitoring as well as for general vibration monitoring of passive primary components.

The monitoring procedure is based on the detection of deviations from reference signatures. For this purpose, a statistical discriminant is used which provides increased sensitivity for the detection of spectral changes. To qualify observed spectral changes, the frequency and amplitude of characteristic resonance are used as additional monitoring parameters.

During the development phase of COMOS, comprehensive tests were carried out to test the suitability and sensitivity of various statistical discriminants or similarity measures to detect spectral changes. Although the discriminants are global measures of spectral change and as such unsensitive, an application to narrow frequency intervals enables some discriminants to detect spectral change sensitively, results in a significant feature extraction and data reduction and allows the integration of results obtained from a detailed correlation analysis. It was found that deterministic influences, such as operational influences, normal changes of vibrational behaviour during a fuel cycle and electrical disturbances might cause false alarms in an on-line system if they are not properly dealt with.

To account for the redundancy of information in different signals and to incorporate the causal relationship between different primary components, a linkage of signatures was introduced. This new approach allows a significant incorporation of a knowledge base using normal computational methods and gives the system diagnostic potential. Furthermore, with this principle and the application of a time criterion, most transient deterministic influences could be coped with.

A modular hardware and software concept is discussed. A high degree of automation and information extraction and a user-friendly representation of measurement results allow utility people to apply early failure detection methods without continuous expert assistance. Finally operational experience and system acceptance is highlighted.

KEYWORDS

PWR's; vibration monitoring; condition monitoring system; primary circuit components; main coolant pumps; shaft rupture; statistical discriminants; signature linkage; spectral change; operational experience; neutron noise; pressure noise; vibration sensors;

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PROBLEM - ORIENTATED VIBRATION MONITORING

Noise analysis techniques provide information about the mechanical integrity of components at any time during the power operation of a plant. Beneath the general information of a potential mechanical deficiency, an observation of the continuing development of this deficiency, the judgement on its severity and consequently the decision if and when the plant operation has to be changed are the main scopes of vibration monitoring. In this context, the time dependence of mechanical degradation is extremely important. Looking at the monitored components of a PWR, two groups can be separated:

- Passive components like vessels, piping systems, pump-housings and steam generators are mainly forced by static loads like pressure, dead weights or internal stresses - the propagation of mechanical defects is extremely slow.

- Active components with a high energy conversion like pump-shafts, turbine rotors or motor-driven valves are primarily loaded by dynamic forces. As the load cycles are extremely high, short-term degradation has to be assumed.

Both types of failure characteristics have to be included in a condition monitoring system with problem-orientated features.

Condition monitoring of passive components

Results of research and development of PWR vibration monitoring meanwhile lead to a certain standard instrumentation for the primary circuit - sensor types and sensor locations are indicated in figure 1. The sensors used are inductive absolute displacement sensors on top of the reactor pressure vessel, relative displacement sensors at the pump housing and on hot leg positions near the steam generators, neutron noise sensors of the ex-core safety instrumentation and piezo-electric pressure sensors located at inlet and outlet pipes. This sensor array can be used for periodical measurements, to analyze the signals for undue deviations. The vibration analysis is based on power spectral densities and correlation functions of selected signal pairs. In order to achieve a complete understanding of these spectra within the frequency range of interest, systematic measuring campaigns during pre-operational and operational plant conditions and the gradual adaption of mechanical and hydrodynamical models was a prerequisite [1]. Together with an excellent cooperation with utilities of various plants (e.g. backfitting of inspection results and the vendor of the plants KWU /2/ (e.g. definition of modal parameters by shaker tests), a detailed interpretation of the signal patterns is available of many PWR's /3/, /4/. Figure 2 highlights the interpretation results for the KKG-PWR.

![Fig.1: Vibration and noise sensor positions in KKG-PWR](image1)

![Fig.2: Interpretation map as a result of baseline analyses](image2)
As per nuclear standard of the Nuclear Safety Standard Commission, periodical measurements of the vibrational behaviour are required at least three times per fuel cycle, to analyse the signals for undue deviations. In the past, additional measurements had been carried out in order to get a better understanding of the long-term behaviour, especially under changed operational conditions.

Operational experience with passive components can be summarized as follows:

- the monitoring is based on frequency deviations of natural frequencies
- the development of mechanical degradation is rather slow, typically in the time span of several weeks to several months (e.g. material fatigue in components, bearings and clappings)
- additional systematic deviations are mainly linked with the fuel cycle time (e.g. fuel assembly spacer grid relaxation, changing boron concentration, stretch-out conditions)
- changes of operational conditions of the plant lead to short-term spectral deviations with timing constants of several minutes up to some hours (e.g. load follow operation, on-line tests of auxiliary systems).

Condition monitoring of active components

Concerning active components, our work is focussed on shaft vibration monitoring of main coolant pumps. As shown in figure 3, several shafts were damaged since 1973 in various plants. In addition to high dynamic load cycles due to the hydraulic forces of the impeller, main reasons for shaft rupture were stress overloads due to wrongly designed circumferential grooves in the lower shaft area, sometimes in combination with residual fabrication stresses or thermal stresses from cool seal water injection. All cracks were positioned below the lower radial bearing. For this reason no severe damage of the gasket group occured. In nearly all cases, the first indications of abnormal shaft operations were seal leakages, accompanied by higher than normal pump shaft vibrations or pump motor vibrations. Unfortunately in most cases the vibrational characteristics were not monitored in a consequent way and in some cases these indications were misunderstood as incorrect measurements or led only to additional balancing of the shaft.

In figure 4 shaft vibration amplitudes, measured in the KKG-PWR before the MCP4 shaft ruptured (4th December 1986, 9.08 a.m.) are shown. A distinct rise in amplitude from 50/100 µm up to 450/500 µm can be seen. The results were obtained from two perpendicular eddy current sensors positioned at the gasket group of MCP 4 (figure 1). The vibrational history of both sensors is reconstructed from broadbanded strip chart recordings. The cross section of the shaft rupture, inserted in figure 4, demonstrates a crack propagating unsymmetrically from an initial crack to the fracture residual. From these facts it can be deduced that the detection efficiency for vibration monitoring would have been excellent. On the other hand, the crack propagation in the final stage (correlated with the rise in vibration amplitude) demonstrates rapid failure development.

**MCP SHAFT DAMAGE**

<table>
<thead>
<tr>
<th>DATE</th>
<th>PLANT</th>
<th>TYPE</th>
<th>EVENT</th>
</tr>
</thead>
<tbody>
<tr>
<td>30. Nov. 1973</td>
<td>SURRY-1</td>
<td>WH-PWR</td>
<td>rupture</td>
</tr>
<tr>
<td>2. June 1981</td>
<td>PRAIRIE ISLAND-2</td>
<td>WH-PWR</td>
<td>300° crack</td>
</tr>
<tr>
<td>30. Jan. 1984</td>
<td>TMI-1</td>
<td>BNW-PWR</td>
<td>50% crack</td>
</tr>
<tr>
<td>18. Sept. 1984</td>
<td>PALUDES</td>
<td>C-E-PWR</td>
<td>Impeller disconnection</td>
</tr>
<tr>
<td>6. May 1985</td>
<td>GÖSGEN</td>
<td>KWU-PWR</td>
<td>rupture</td>
</tr>
<tr>
<td>1. Jan. 1986</td>
<td>CRYSTAL RIVER-3</td>
<td>BNW-PWR</td>
<td>rupture</td>
</tr>
</tbody>
</table>

Fig. 3: Shaft damage occurrences of several PWRs

Fig. 4: Shaft vibration amplitudes measured in KKG-PWR before shaft rupture
It is evident from these facts, that there is a real possibility to identify the presence of a crack in a radially loaded vertical main coolant pump by means of vibration analysis techniques without resorting to extensive trial and error efforts. To provide proper interpretation and avoid misleading conclusions, in-depth knowledge of the shaft vibration has to be gathered. Consequently it is necessary to make maximum use of all available data. This includes a combined vibration analysis of pump shaft and pump casing and the investigation of operational influences on the vibration signatures. Therefore a frequency analysis of the signals is needed, to monitor the peaks corresponding to shaft rotation (1X) and higher harmonics (2X, 3X, 4X...) in a detailed trend analysis.

Experience regarding shaft rupture at main coolant pumps can be summarized:

- the monitoring is based on amplitude trend analysis of rotation specific frequency components
- crack propagation has been observed to develop in the time span of several hours to several months.
- changes of operational parameters (e.g. primary pressure, primary temperature, sealing water pressure, sealing water temperature) may influence the shaft vibration signals in a strong way.

THE USE OF PATTERN RECOGNITION

Due to the possibility of rapid failure development, it became necessary to support the monitoring procedure using on-line mathematical methods in such a way that component status is presented to the operator in a transparent way.

To solve pattern recognition problems, many different mathematical techniques can be used /5/. These techniques can be grouped into two general categories, namely the statistical (decision-theoretic) and the syntactic approach. In the statistical approach, different methods can be used: Non-parametric decision-theoretic classification, Bayes (parametric) classification and cluster analysis.

The type of pattern recognition problem normally determines which approach should be used. To name a few aspects, the following should be considered:

- the nature of the process, events or features to be classified (stochastic and/or deterministic)
- the information provided by feature measurement and the cost of taking it
- the importance of structural information which describes the pattern
- the complexity of the pattern
- whether the number of pattern classes are precisely known
- whether or not reference vectors can be selected appropriately.

In our case the problem of pattern recognition denotes a discrimination or classification of the vibrational behaviour of primary components of a nuclear power plant. The sensor location and signal array used to monitor the process has been shown in figure 1.

The process of which these noise signals are the output, has been treated as a purely stochastic process in some applications of pattern recognition. In continuous on-line monitoring applications however, this is not a valid assumption. Many deterministic influences may cause false alarms if they are not properly dealt with. These influences may be electrical disturbances, operational influences or normal changes of vibrational behaviour during a fuel cycle.

This problem may be solved by the inclusion of additional plant parameters for the monitoring procedure, resulting in a number of disadvantages: Uncertainly as to which parameters should be included may result in the inclusion of many parameters. This might increase the complexity of the monitoring problem, require additional cabling or hardware costs.

We suggest an alternative approach which minimizes the need for additional plant parameters and may be seen as a combination of pattern recognition techniques and a knowledge based approach.
In minimum distance classification and nearest neighbour classification the distances between the input pattern and a set of reference vectors in the feature space is used as the classification criterion. For pattern recognition problems with the number of classes not precisely known, cluster analysis has been regarded as a practically attractive approach. Cluster analysis may be characterized by the use of similarity or dissimilarity measures between pattern samples in the feature space. Various similarity and distance measures have been suggested as classification criteria. A similarity measure quantifies the similarity or dissimilarity between two reference patterns.

Let $X_i = (x_{i1}, x_{i2}, \ldots, x_{in})$, $i=1,2,\ldots,m$ be a number of features in the frequency domain ($X_i$ may be APSD, CPSD, Coherence or the phase calculated for a particular noise signal or signal combination). Let $x_{ij}$ be the value (e.g. $V^2/\text{Hz}$) of spectral line j. Let $X_p$ refer to a reference pattern.

During the development phase of COMOS many different similarity measures (discriminants) were tested and compared to ascertain their suitability and sensitivity to detect spectral changes (figure 5). Tests were performed for linear as well as logarithmic spectra. The discriminants suggested by Piety were also compared /6/.

The goals of these tests were:
- to obtain further data reduction and feature extraction. As few as possible discriminants were sought to describe spectral changes uniquely
- to increase the sensitivity with which spectral changes can be detected
- simplicity and calculability
- to study the time dependent vibrational behaviour of some primary components during different operating conditions
- to study the influence of operational changes on the vibrational behaviour of selected primary components.

These tests were carried out in four phases /7/:

a) during steady state conditions
b) with artificial spectral deviations
c) with real spectral changes
d) during stretch-out conditions, load-follow operations and other operational influences

to a)

A 12 hour tape recording of 14 noise signals was made at the GKN 3 loop 855 MWe PWR of the KWU. The recording was made when steady state conditions prevailed. A number of plant parameters were recorded simultaneously. The discriminants mentioned were applied to a number of selected frequency intervals. Some discriminants indicated a small spectral deviation in the frequency band corresponding to the subharmonic shaft vibration of MCP1 (characteristic to Andritz designed MCP's) and at a time during which a small change in the differential pressure of MCP1 was detected. It was found that too little data was available to adequately test any assumptions regarding the statistical nature of the data.

to b)

Spectral changes can be characterized by
- resonance frequency changes
- peak amplitude changes
- peak shape changes (e.g. peak width changes which might indicate a change of damping behaviour of mechanical components)
- change of background noise level
- any combination of the previous
with a white noise background peak frequency, amplitude and width changes (as well as a few combinations of these changes) were simulated for different peak amplitude to background noise ratios. To obtain a more realistic picture, a noise signal (from the 12 hour tape recording) in which no conspicuous spectral changes had been observed, was used to simulate spectral changes in

- a large artificial peak with and without background noise
- a small artificial peak with background noise and
- a small artificial peak with background noise and close to a real peak

as shown in figure 6. The analyzed frequency intervals are indicated. In figure 7 these spectral changes are shown with two examples of corresponding discriminant behaviour. Only the last six steady-state spectra are shown, following spectra
containing simulated peak frequency changes (-10/16 Hz to +10/16 Hz in steps of 1/16 Hz), peak amplitude changes (-20% to +20% in steps of 2%) and peak width changes (half peak width to double peak width in twenty steps). In figure 8 three more discriminants are shown. The dots represent calculated discriminants for steady-state spectra, following discriminants calculated for peak frequency, amplitude and width changes (indicated with +, * and #) as discussed above.

to c)

The discriminants were also applied to some spectra contained in the GRS data base. For this purpose, four noise signals (A2, R1R, W1R and XIU) of the GKN nuclear power plant were used (sensor location is similar to that of KKG-PWR as shown in figure 1). No indication of statistical fluctuations of calculated discriminants can be given as only 15 measurements during 5 fuel cycles were available. Four frequency intervals were chosen for calculation of discriminants:

- 0 to 50 Hz
- around 3 Hz (MCP pendular vibration)
- around 25 Hz (MCP rotation specific)
- around 8 Hz (Core Barrel pendular vibration)

as indicated in figure 9 (top, left). Examples of two corresponding discriminants as calculated for the frequency intervals containing the Core Barrel pendular vibrations, are given in figure 9. Both discriminants indicate two deviations for noise signals A2 and XIU. These deviations correspond to 80% power operation and measurement shortly after two-loop operation. (In both cases the corresponding spectral change can be explained.) The same effect was not seen in the R1R noise signal due to sensor location.

![Fig. 9: Test of statistical discriminants with real spectral changes](image)

![Fig. 10: Test of statistical discriminants during stretch-out conditions and other operational influences](image)

to d)

During May 1987, a 448 hour tape recording of 14 noise signals was made at the GKN nuclear power plant. A number of plant parameters were recorded simultaneously, such as reactor power and MCP3 seal water pressure (figure 10, top). The time span included normal plant operation (approximately 112 hours), stretch out operation and load follow operation. The discriminants were applied to a number of selected frequency intervals in all recorded noise signals. For each signal approximately 2700 spectra were calculated. In figure 10 two examples are given...
of the correlation discriminant as calculated for two frequency intervals containing the MCP3 subharmonic shaft vibration and the MCP1 pendular vibration. A number of operational influences are indicated. Most discriminant deviations were correlated with operational influences or electrical disturbances and had a short duration (except for load-follow operation and the influence of stretch-out operation).

Discussion

From these tests, the following results, relevant for continuous monitoring purposes, can be deduced:

- The computational effort necessary to calculate discriminants varies considerably. Computationally intensive discriminants, such as the Benzecri and Mahalanobis metrics, did not offer a marked improvement in sensitivity.

- The discriminants are global measures of spectral changes and spectral changes can not be detected uniquely if they are applied to wide frequency intervals containing several peaks. In this case, simultaneous spectral changes (in more than one peak and/or background noise) might complicate the qualification of indicated deviations. It is desirable to apply discriminants to narrow frequency intervals containing single frequency resonances. (If two frequency resonances overlap in a monitored frequency interval, an indicated spectral change can be qualified using redundant information as discussed in the next section.)

- The sensitivity of the various discriminants to detect spectral changes varies considerably. Some discriminants are sensitive only to positive or negative peak frequency changes, others are more sensitive to amplitude or peak shape changes. These discriminants can only be used in combination with others to detect spectral changes uniquely. A few discriminants (such as the discriminants listed in figure 5) can detect spectral changes uniquely with sufficient sensitivity.

- The correlation metric was chosen to be incorporated in COMOS as it was found to be sufficiently sensitive to all possible spectral changes in narrow monitored frequency bands. The correlation metric quantifies spectral change and maps spectral change to a real interval [0,1] where values close to 1 indicate spectra similar to baselines and values close to 0 indicate spectra very dissimilar to baselines in the monitored frequency intervals.

THE INCORPORATION OF EXISTING KNOWLEDGE: INFORMATION LINKAGE

Although a purely statistical approach may provide a sound statistical description of the problem in ideal circumstances, most pattern recognition methods have limited learning capabilities. Normally a learning phase is introduced at the beginning of each fuel cycle. Some algorithms have updating learning capabilities which can be applied to ideal processes.

A knowledge-based approach should account for physically well-founded causal relationships between the vibrational behaviour of different mechanical components. The following information should be incorporated in an automated on-line surveillance system:

- The fact that we use different signal types containing redundant information. Some sensors "see" the same effect with a sensitivity according to sensor location and type.

- A complete interpretation of different resonance frequencies calculated from different noise signals. This information is normally the result of a correlation analysis of a well-instrumented prototype system, shaker tests, mechanical models or model calculations.

- The different time behaviour of mechanical degradation and disturbances.

- Concerning failure development experience gathered in some plants may be applied to other plants qualitatively in justified cases. This may often be done for plants with similar structural design in similar operational states. Several well documented mechanical failures (which had been measured, predicted or analysed using noise methods) can be found in the literature /4/.

- Postulated mechanical failure has been modelled. Results may be applied qualitatively in justified cases /8/.
- Some spectral changes are unambiguously indicative of operational influences. This is true for the resonances corresponding to standing pressure waves as can be seen in spectra belonging to the pressure noise signals as well as in some noise signals belonging to absolute or relative displacement sensors.

It has been shown that spectral changes can be detected sensitively using a sensitive discriminant. In order to qualify the spectral changes as indicated by such a discriminant, it makes sense to use the frequency and maximum amplitude belonging to each resonance peak as additional monitoring parameters. Furthermore, in some cases monitoring experience is restricted to amplitude deviations (e.g. MCP shaft monitoring) or frequency deviations of certain frequency peaks. Alert levels can be set taking into account the statistical fluctuations of the monitoring parameters and/or using heuristic information.

By allowing a knowledge-based choice of monitored frequency bands, the information obtained from detailed correlation analyses can be incorporated. Therefore resonance peaks with physical significance can be used in the monitoring procedure. Once the monitored peaks have been identified, more information is often available such as time behaviour (for normal operation and sometimes even for failure development) of components to which chosen resonance frequencies belong. Furthermore, information regarding signal stability can be accounted for. In some noise signals, instability or disturbances in some frequency bands make resonance peaks unsuitable for monitoring purposes.

To account for the redundancy of information in different signals, and to incorporate the causal relationship between different primary components, it was necessary to introduce a linkage of information contained in various resonance frequencies.

To allow a general application of the principle, it was necessary to allow a linkage of spectral changes between any two of the chosen resonance frequencies belonging to any of the noise signals of the signal array. In figure 11 a block schematic is shown. This may be seen as a symmetric decision matrix and can be obtained for any of the monitoring parameters discriminant, frequency or amplitude. The actual results are displayed on the diagonal using pre-defined symbols. It is shown that peak changes can be compared generally and that comparison of peak changes in different signals can be found in specific rows and columns in this matrix.

A linkage of peak changes has to be defined and for this purpose pre-defined rules can be used specifying the required linkage. These rules may be defined using pre-defined symbols. This procedure allows the use of normal computational methods to accept and apply the rules flexibly.

Fig. 11: General principle of information linkage

Fig. 12: Hold down spring degradation as an example of information linkage
In addition to the linkage of information, a time criterion can be specified. The time criterion requires that an alert level has to be exceeded for a pre-defined timespan before it is taken as a true indication of alert level excession.

To summarize, correct application of these principles, results in the following:

- Transient electrical disturbances can be coped with. Most transient electrical disturbances have one or more of the following characteristics: They have a short duration (mostly seconds), can often be seen in more than one monitored frequency band simultaneously and are often found in more than one signature simultaneously. These influences are most often suppressed by the time criterion.
- Transient operational influences are dealt with. Most transient operational influences can be seen in certain peaks as discussed above.
- Normal mechanical change during a fuel cycle can be dealt with and
- Experience as described above may be incorporated.

To illustrate the application of signature linkage, an example is shown in figure 12. Hold down spring degradation has been chosen /4/. A typical set of rules describing this situation for normal plant operation could be (within the framework of the given signal array):

IF there had been no transient electrical disturbances
AND IF there had been no transient operational disturbances or influences as can be deduced from pressure signatures
AND IF the resonance frequency corresponding to core barrel pendular vibration has exceeded its alert level and the frequency decreased for noise signatures belonging to the absolute displacement and excore neutron sensors
AND IF no spectral change can be seen in any relative displacement spectrum in the frequency band corresponding to the frequency band mentioned above
THEN Detailed investigations regarding hold down springs should be started.

(Assume for simplicity that only 2 absolute displacement signatures, one pressure signature and one excore neutron noise signature are available (as in figure 12) having signal numbers 1, 2, 3, 4. Assume that the peaks corresponding to core barrel pendular vibration, standing wave in inlet/outlet piping have peak numbers 1, 2, 3 and that a time criterion has been fulfilled.)

Let Sij denote peak number j of signal number i. D/F/A refers to Discriminant/Frequency/Amplitude for corresponding peaks. Applying the pre-defined symbols, the above rule may be written compactly as follows:

Rule1$=\text{D/(S}_{11}^* A S_{21}^* A S_{32}^* A S_{33}^* A S_{41}^* A F/(S_{11}^v A S_{21}^v)}$ : special attention regarding hold down springs necessary

Regarding MCP shaft monitoring, requiring alert level excession for corresponding peaks of proximity sensors measuring in two directions may be written: (assume signal number 1/2 for signatures from noise signals W1X/W1Y and let rotation specific frequency peaks have peak numbers 1, 2, 3, 4. Again it is assumed that the time criterion has been fulfilled.)

Rule2$=\text{A/(S}_{11}^* A S_{21}^*)O(S_{12}^* A S_{22}^*)O(S_{13}^* A S_{23}^*)O(S_{14}^* A S_{24}^*)$ : special attention regarding MCP shaft required

It is realised that the applicability of specified rules depends on the choice of alert levels, especially where "And"-restrictions apply. Therefore it is suggested that "And"-restrictions may be applied in cases where signature changes can clearly be distinguished from normal background fluctuations. In other cases, "And"-restrictions should not be applied strictly and corresponding rules should merely be seen as a diagnostic aid.
SYSTEM HARDWARE CONCEPT

For a detailed correlation analysis of reactor noise signals, it is normally necessary to obtain APSD's of all signals, as well as CPSD's of many relevant signal combinations. It is advisable to make this analysis from simultaneously recorded noise signals.

For on-line monitoring purposes, the information obtained from such a correlation analysis can be applied, and the monitoring can be restricted to the calculation of APSD's for all noise signals and CPSD's for a few essential signal combinations. Therefore a 2-channel system can be used. Since the excess of alert levels is linked with a time criterium as well as other criteria as discussed in the previous section, a 2-channel system offers no draw-back.

The COMOS hardware configuration is shown in figure 13 and figure 14. A 32-bit microcomputer controls two 16-channel multiplexers, a two channel frequency analyser, a failure indication module and output peripherals. This configuration allows calculation of noise spectra for noise signals (two at a time) in a pre-defined sequence and cycle time.

Fig. 13: Block schematic of COMOS hardware configuration

Fig. 14: COMOS with front door open and pulled out keyboard

This hardware configuration offers a few advantages:

- Except when receiving data from the analyser, the computer is free to perform other tasks such as graphic displays, trend plots or supervised learning.
- The number and range of monitored frequency bands, the monitoring parameters (discriminant, frequency, amplitude or other monitoring parameters applicable in the frequency domain) as well as the linkage of information as described, may be specified freely. Noise signals can be grouped in a logical way and measurement repetition can be adapted to the specific problem. This flexibility gives the system a problem-orientated character.
- The applicability of the system is enhanced by the frequency analyser features.
- A frequency analyser and some output peripherals are normally available for noise analysis in nuclear power plants. Integration of these components reduces hardware costs. A modular configuration allows decoupling of these components for other purposes, and allows updating with new developments such as the use of array processors, etc.
SYSTEM SOFTWARE CONCEPT

To address the different time behaviour of mechanical degradation in passive components and crack propagation in MCP’s, the software was developed for operation in two modes. Some of the features of each mode is listed in figure 15. In mode 1 (always active except during 3 hours a week) the MCP shaft crack monitoring is addressed. For each of four MCP’s, four noise signals are available: two proximity sensors and two relative displacement sensors. In mode 2 (active only 3 hours a week when steady-state conditions prevail) the general problem of passive component surveillance is addressed. For this purpose the mentioned noise signals are available, as well as two relative displacement signals, 2 absolute displacement signals, 2 pressure signals and 4 excore neutron noise signals (figure 1).

Three vital aspects have to be considered in an on-line noise analysis system:

- a high degree of automation
- the human interface
- information extraction and condensing.

![Fig. 15: COMOS features](image)

![Fig. 16: Menue driven software for additional information output](image)

These features are intended to allow utility people to apply early failure detection methods without continuous expert assistance.

High degree of automation

COMOS requires no human intervention for its normal operation. It has the following automation features:

- upon power on or after power failure the system boots itself, initializes peripherals and continues operation
- data gathering and computation of monitoring parameters are done automatically
- all vital outputs are done automatically:
  - a daily Status Plot for all four MCP’s
  - documentation of peripheral component failure
  - documentation of alert level excession
  - continuous CRT display of actual MCP monitoring results
  - a weekly printout of results following mode 2 operation
  - a control room indication of hardware failure or signal alert level excession (for mode 2)
- mode 2 (general vibration monitoring) is started automatically once a week and upon completion of the measuring cycle, the system resumes mode 1 operation.

The human interface

Except for the information contained in the mentioned outputs, additional information may be obtained using menu-driven software (figure 16) which does not interfere with the measuring process and calculation. Fast colour graphics increases user friendliness and alleviates the task of information condensing and interpretation. A built-in help function for each menu provides a short description of softkeys. The keyboard is locked (except for softkeys) to avoid intervention with normal operation due to typing faults.
Information extraction and condensing

Pattern recognition has the advantage that it achieves a data reduction and it is often argued that only conspicuous spectra need to be stored during the monitoring process. A proper judgement of the development of mechanical degradation requires the observation of the time behaviour of such a development (trend) within a time span which is relevant for the monitored components. Therefore the system was layed out to provide a past history of spectra and monitoring parameters of one month for mode 1 and a complete fuel cycle for mode 2. Only after these timespans are records overwritten.

For mode 1, the relevant information contained in 90 000 averaged spectra per year are displayed on 4 trend plots per month. For mode 2, the relevant information contained in 3 000 spectra of 32 noise signals are condensed to 8 trend plots or 2 status plots per year.

In each mode, the software operates on two levels:

On the first level, running with the highest priority, the system performs its tasks as described above. On the second level, running with lower priority, a number of menu-driven functions are available (figure 16). These functions are intended to allow the operator to obtain additional information beneath the mentioned automatic outputs. Some of the included possibilities are shown and discussed in the next section.

The software includes subroutines to allow a learning phase when required. A learning phase is normally introduced at the beginning of each fuel cycle or when new component operating conditions prevail (e.g. after changing of bearings, seals, etc.) and only in justified cases. During the learning phase, baseline spectra are calculated, and an indication is given of statistical fluctuations of monitoring parameters. Alert levels and other parameters relevant for the monitoring process may be specified. Although unsupervised learning is allowed, supervised learning is favoured and can easily be applied due to system layout. A learning phase may be started for any number of available noise signals.

OPERATIONAL EXPERIENCE

Operational experience has been made with two systems, installed in the PWR's GKN and KKG. Operation of GKN-COMOS started on the 12th December 1986 (eight days after the pump shaft ruptured in Grafenrheinfeld), and continued until the end of the eleventh fuel cycle seven months later. The KKG-COMOS was installed on the 24th May 1987 and the first test-phase was completed with plant shut down (for maintenance) five weeks later. The prototype COMOS in GKN is running with a software concept for active components, comparable with MODE 1. A complete adaptation to the KKG-standard will be realized before the end of the year. In GKN 62 000 spectra (MODE 1) and in KKG 25 000 spectra (MODE 1) and 800 spectra (MODE 2) have been gathered, stored and analyzed until September 1987.

Condition monitoring of active components (KKG - MODE 1)

The monitoring is based on shaft vibration spectra in the frequency range 0-100 Hz. Additional information, regarding pump casing vibration, can be derived from relative displacement spectra. Monitoring is restricted to the amplitude behaviour of the dominant peaks in shaft vibration spectra (figure 17). These peaks are due to MCP shaft rotation (24.8 Hz), as well as higher harmonics at 49.7 Hz, 74.5 Hz and 99.3 Hz. The shaft vibration spectrum in figure 17 gives an example of the 24-hours amplitude trend for these peaks. Trend curves and associated peaks are marked in corresponding colours. The peak amplitudes are normalized to a reference state and deviations are given in percentage. To improve the understanding of observed amplitude trends, a combination of all relevant signals of one main coolant pump is condensed to a standard display (figure 18): 24-hours amplitude trend of various frequency peaks in four different signals can be compared. In the relative displacement spectra, not all desired peaks could be used in the monitoring procedure (figure 18, lower part): Some of the higher harmonics are covered by structural resonances of the piping system or the signal-to-noise-ratio is insufficient. These peaks have been neglected.
Every observed deviation could be treated in the same way:

- the 'Standard Display' enables a comparison of peak amplitude behaviour of all monitored peaks of 4 monitored signals of one MCP. The time of occurrence is given
- using the 'Status Plot' to obtain information as above and to allow a comparison between all MCPs
- using the 'Graphic Menue' for detailed spectral analysis
- using the 'Long Term Trend Menue' to obtain a general view over a longer time span.

The experience can be summarized in the following way: Shaft vibration was found to be stable (figure 18, upper part). In the shaft vibration spectra, amplitude trends (of the peaks at 50 Hz and 75 Hz in two shaft vibration signals) was observed. These trends were not found in redundant shaft vibration spectra (x- and y-directions) and the absolute value of the deviation was between 4 μm and 8 μm (compared to an amplitude of 79 μm to 100 μm of the 25 Hz peak). All trends disappeared after a few days except in one case which was due to signal conditioning.

Condition monitoring of passive components (KKG - MODE 2)

Based on a fairly complete interpretation of the primary system vibration characteristics, a set of structural resonances was selected for condition monitoring: Pendular- and vertical reactor pressure vessel vibration, beam mode and shell mode vibration of the core barrel, beam mode vibration of the secondary core support structure and of fuel assemblies, pendular- and vertical vibration of all main coolant pumps and vertical vibration of the steam generators. Monitored frequency intervals are indicated in figure 2 (†). The most important fluid-resonances in inlet- and outlet-pipes were also included in the set of monitored frequency bands, to obtain information about pressure and temperature conditions in the primary coolant.

Figure 19 exemplifies fluid-resonances in an outlet pressure spectrum. As the plant was running under stretch out conditions during the last two weeks of the fuel cycle, standing wave resonances showed a certain deviation corresponding to local inlet/outlet coolant temperature changes. Corresponding trends can be observed in discriminant behaviour; the correlation discriminant decreases from 100 per cent to 72 per cent. In the Standard Display in figure 20, two pressure spectra (P3A, P3E) and two RPV vibration spectra (A1, A4) are included. As the stretch out conditions of the plant reduced the power-level of the plant by only 15%, the influence on structural resonances is weak. The corresponding 4 weeks discriminant trends of RPV-vibration resonances show a fairly normal behaviour.
Separation of mechanical and operational influences (GKN-data)

In figure 21 and 22 two examples are given of results obtained with COMOS in GKN. The correlation discriminant, calculated for a number of frequency intervals in a few noise signals are shown. The use of signal-linkage is exemplified. In figure 21 a similar discriminant behaviour for two redundant noise signals (R1D and R1R) calculated for the MCP1 pendular resonance can be observed. This behaviour is due to frequency shifts of the MCP1 pendular resonance. (During the last five years a frequency shift of this resonance from 3.9 Hz to 3.6 Hz has been observed. As standard measuring campaigns took place every three to five months, the effect could not be studied in detail.) This is a real mechanical effect because

- the time criterion was fulfilled, and
- the effect could be seen in both measuring directions (the R1D and R1R relative displacement sensors have a similar sensitivity to MCP pendular vibration), and
- the effect could not be seen in corresponding noise signals positioned in other loops, or in other frequency intervals of the same signal. (The effect could not be seen in W1R and W1D spectra, but these signals are less sensitive to pump casing vibration.)
A separation of operational influences may be more complicated. An example is given of subsynchronous shaft vibration, which is very typical for the GKN-MCP’s. A power level change on the 10th April, 1987, could be seen in various noise signatures. Figure 22 shows peaks in the discriminant trend in shaft and casing vibration signals of one pump (W3D, W3R, R3R) as well as in other pumps (e.g. R2D of MCP-2). In other frequency ranges similar behaviour was found (e.g. the fluid resonance at 37 Hz in sensor R2D). Other operational influences may be seen in signatures of only one pump (e.g. MCP3 sensors during manual coolant temperature control) or only in one sensor-type of one pump (e.g. MCP3 shaft sensors during operation of a second seal water pump). Shaft bearing influences are pump specific and were only seen in signatures of the corresponding MCP. Both examples underline the high potential of signal linkage to support complicated failure diagnostics and to suppress false alarms.

FUTURE PERSPECTIVES

It is planned to have three identical systems running before the end of 1987:
- an update of the GKN-COMOS is under preparation
- the KKG-COMOS is in operation since May 1987
- a third system will start operation in KKI-II (Isar) during November 1987.

All three systems will operate with hardware and software described in this paper.

The intensive and sometimes complicated task of signature analysis is substituted by the interpretation of trends. This interpretation is alleviated by the application of pattern recognition and the integration of existing knowledge and experience. Measurement results are presented in such a way that the operator can obtain over-view and pre-view of deviations or trends. These facts lead to a higher acceptance of early failure detection methods by utility people and enable the use of the condition monitoring system without continuous expert assistance. It is intended to increase this acceptance by backfitting experience obtained with all systems.

Other future perspectives include:
- the integration of more rules, specifying a signature linkage, to increase the diagnostic possibilities of the system
- a continued study of the time dependance of the vibrational behaviour of primary components as more data becomes available
- an investigation of the statistical implications of signature linkage
- a wider application of noise analysis methods such as sensor or signal conditioning monitoring and thermohydraulic monitoring.

The authors believe that the approach to vibration monitoring and anomaly detection described in this paper has unique features and a high potential to solve a complicated problem.

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FAST FLUX TEST FACILITY NOISE DATA MANAGEMENT

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Abstract - An extensive collection of spectra from an automated data collection system at the Fast Flux Test Facility has features from neutron data extracted and managed by database software. Inquiriy techniques, including screening, applied to database results show the influences of control rods on wideband noise and, more generally, abilities to detect diverse types of off-normal noise. Uncovering a temporary 0.1-Hz resonance shift gave additional diagnostic information on a 13-Hz mechanical motion characterized by the interference of two resonances. The latter phenomenon is discussed generally for possible application to other reactor types.

Keywords - Fast reactor; LMFBR neutron noise analysis; reactor surveillance; control rod noise; database software; spectral resonances; Fast Flux Test Facility; interfering resonances.

1. INTRODUCTION
The Oak Ridge National Laboratory (ORNL), in cooperation with the staff of the Fast Flux Test Facility (FFTF), acquires neutron and process noise spectra on the FFTF. An automated data-gathering system (Mullens et al., 1984) is used on this reactor, which is a U.S. government-owned, 400-MW(t) sodium-cooled fast reactor used for test purposes. Objectives of ORNL's noise data collection program include demonstrating automated surveillance.

This system stores spectra and cross spectra on a hard disk. The data accumulation capacity of this system is equivalent to 3840 graphs of spectral functions a day for almost every day of the year. "Eyeballing" by noise specialists is clearly limited to spot-checking. The purpose of the present investigation is to implement a much-needed tool--database management software--to thoroughly examine all data collected. Examples below show some physical insights gained in trial applications of this method, although these do not represent the primary objective of this paper.

2. DATABASE METHODOLOGY
The problem being solved here is how to utilize large masses of data stored on hard disk to give very specific types of information in answer to narrow and precisely formulated questions. In many industries this problem has been solved by the employment of sophisticated database management computer programs. Following these precedents, a commercial software package (Microrim, 1986) was selected for this research. User-friendly commands allow data-moving manipulations as well as arithmetic and logical operations.

It was recognized at the outset that computing efficiencies would be achieved by storing spectral features rather than numerous spectral points in the database. Fig. 1 indicates that more than one such feature can be extracted and stored in a master table. The first feature extraction algorithm used in this application was spectra integrated from 0.47 Hz to 5.17 Hz to obtain rms values; the second was a peak extraction algorithm to obtain resonant frequencies. Appropriate identifying conditions--control rod configuration and type of detector used--are also entered in the database. Managed by the software, disk file to disk file transfers occur throughout the processes of Fig. 1.

*Operated by Martin Marietta energy Systems, Inc. for the U.S. Department of Energy under Contract No. DE-AC05-84OR214000.
Each master table in the database contains at least two columns with related information. The rows correspond to different features (sequentially acquired) at associated conditions. Specific examples are

- a. Columns for a rms table: an identification number (ID) such as time or test number, the rms for channel 1, and the rms for channel 2.
- b. Columns for a resonances table of channel 1: ID, resonant frequency, and spectral value at this resonance.
- c. Columns for a conditions table: ID, control rod number, rod position, type of detector used for channel 1 in spectral analysis, and type of detector used for channel 2.

The database software allows selectively combining judiciously chosen portions of these tables into master tables having specialized purposes by taking advantage of commands to execute mergings and removals in accordance with logical or arithmetic formulated instructions. For example, one combines rms and conditions tables, then into this new table dc values belonging to the conditions are added; this results in the ability to have a calculated column, nrms = rms/dc value.

Fig. 1 indicates that a master table of data can be subdivided into special smaller tables to which specific inquiries are more efficiently directed. In an example presented in the next section, a specific neuron detector within a specific range of rod configurations is of interest. The reduced table satisfying these conditions would be extracted from the master table by omitting unwanted sensors and rod configurations. Specific inquiries or screenings can be handled using the reduced table.

3. CONTROL ROD EFFECTS

This section presents an example of the application of this database management technique to data from a special test performed by the FFTF staff in January 1987. In this test one particular control rod was inserted to various depths in the core, while the remaining rods were banked together at a more withdrawn position such that reactor power remained constant throughout the test. Then another particular rod was selected for various insertions. With 45 min spent at each of many configurations, there was ample time for spectral analysis. Three 15-min analyses were performed at each configuration using three low-level flux monitors (LLMs) located in the vessel above the core and three compensated ion chambers (CICs) located outside the vessel. Three days of such data collection yielded important information on how the rms value was influenced by rod configurations.

Obtaining a plot for a particular detector and the position of a particular rod (one of the lines in Fig. 2) is simply a matter of extracting an appropriate reduced table from a suitably constructed master table. The columns of the reduced table were ID, rod number, rod position, detector number, and nrms value. Sections of this table are plotted in Fig. 2.
Fig. 2. The nrmvs value of various neutron detectors versus rod position in three separate tests inserting rod 4, 5, or 9. Regarding rod insertion distances, 36 in. is fully withdrawn.

The insights provided by Fig. 2 on how the control rods influence noise corroborate those of earlier studies (Varnes et al., 1984; Thie et al., 1986):

1. Noise is reduced as a rod is withdrawn toward its full out position (36 in.), since there is less inserted reactivity to be laterally moving within the mechanical guide.
2. In the frequency range selected, 0.47 to 5.17 Hz, the CICs observe this reactivity-induced noise best for the "quiet" rods 4 and 5, because the LLMs observed additional noise from other sources.
3. The rod at position 9 is by far the noisiest, apparently having more freedom of random lateral motion or having more excitation.

The phenomenon here is well understood (Thie et al., 1986) in terms of the reactivity worth of the inserted rod and the mechanical excitation unique to its position.

The ability of the analysis methodology used here to pick out an anomaly is shown in Fig. 3. Since many nrmvs values were obtained at a given rod configuration, it was possible to calculate the standard deviations of these values. A high standard deviation for LLM2 appeared for 0.5 h on January 12, 1987. Its spectrum, shown in Fig. 4, shows the temporary addition of a wideband noise whose origin has yet to be identified. However, no anomalous behavior was observed in its dc value.

4. RESONANCES

One of the master tables constructed contained all of the resonances detected by a peak-seeking algorithm. This algorithm defined a resonance as occurring when there was a reversal in an upward trend of spectral value versus increasing frequency. With spectral statistical accuracy in mind, this reversal needed to be at least a 0.8 factor from the peak attained. Using data obtained from spectra measured during various control rod configurations, 597 rows (stemming from 593 different resonant frequencies) and 6 columns resulted. The columns are ID, resonant frequency, spectral value at resonance, rod number, rod position, and neutron detector used.

Due to statistical uncertainties in measuring a true physical resonant frequency, the resonances fell into ranges clustered about central values. Table 1 shows these central values for the most common resonances. The rich resonant structure of the LLMs compared to that of the CICs is obvious.
Fig. 3.  Standard deviations of sets of nrm3 values for the rod positions of Fig. 2.  For a set of three LLFMs or three CICs, only the largest is plotted; an exception is LLFM2's anomalously large standard deviation being plotted separately from the larger of the other two when rod 9 was at 22 in.

Fig. 4.  Wideband additive noise appearing in LLMF2 during a 0.5-h departure from normal behavior on January 12, 1987.

In one investigation the master table was "windowed" to extract small tables defined by a particular rod being inserted and the frequency range restricted to that of a cluster. The results were that no resonance in Table 1 exhibited any statistically detectable frequency changes due to rod position. With rod configurations now ignored, Fig. 5 was constructed; it is essentially a histogram of all of the spectral peaks of a particular detector. Some occasionally occurring resonances in CIC2 are evident. The number of these rarities for CIC2 exceeds the number of rarities for all other detectors put together. This is an unexplained phenomenon associated with CIC2.

It was seen above (in spotting a nrm3 anomaly) that the large amount of data available enables one to examine statistical measures of quantities. This made possible another investigation: to investigate in a sensitive manner whether individual resonance measurements
Table 1. Central values of peaks in clustered groups of values for the various detectors using data from all PSDs at all rod configurations.*

<table>
<thead>
<tr>
<th>Peak, Hz</th>
<th>LLF1</th>
<th>LLF2</th>
<th>LLF3</th>
<th>CIC1</th>
<th>CIC2</th>
<th>CIC3</th>
</tr>
</thead>
<tbody>
<tr>
<td>7.5</td>
<td>x</td>
<td>x</td>
<td>x</td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>11.</td>
<td>x</td>
<td>x</td>
<td>x</td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>13.25</td>
<td>x</td>
<td>x</td>
<td>x</td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>15.35</td>
<td>x</td>
<td>x</td>
<td>x</td>
<td>x</td>
<td>x</td>
<td>x</td>
</tr>
<tr>
<td>18.2</td>
<td></td>
<td></td>
<td></td>
<td>x</td>
<td></td>
<td></td>
</tr>
<tr>
<td>18.8</td>
<td></td>
<td></td>
<td>x</td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>19.25</td>
<td>x</td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>21.6</td>
<td>x</td>
<td>x</td>
<td>x</td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>23.8</td>
<td>x</td>
<td>x</td>
<td>x</td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>24.4</td>
<td></td>
<td>x</td>
<td>x</td>
<td></td>
<td></td>
<td></td>
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<tr>
<td>25.25</td>
<td>x</td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>26.8</td>
<td>x</td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
</tr>
</tbody>
</table>

*In comparing Fig. 4 with the LLF1 column, bearing in mind that a \( \log_{10} \) reversal at a peak is required, most of these resonances are obvious. Weak resonances, meeting the reversal criterion only sometimes, are not seen in all spectra.

Fig. 5. Density of the spectral resonances that appear in the power spectral densities of the CICs. Bars of a given type essentially represent a histogram of the occurrences of resonances.

are statistically likely in a cluster. Screenings were performed in which small tables were extracted from the master table using all of the latter’s columns but including row-restricting commands such as "project small table from master table using all where ch equals 2 and frequency >13 and frequency <13.5."

The arithmetic/logical instruction at the end of this command selects only rows involving the LLF1 with peaks in the 13- to 13.5-Hz range. One more simple command asking for calculations of basic statistics of this small table’s frequency column gives the following results:

- Count = 30
- Minimum = 13.14
- Maximum = 13.33
- Average = 13.28
- Standard deviation = 0.036
(Note that peaks are located with more significant figures than the spectral resolution of the analysis would dictate, as this is the result of an interpolation feature of the algorithm used.) This particular example is shown because, unlike other such tabulations, the resonant frequency departs anomalously from its average: a single spectrum having 13.14 Hz departs 4 standard deviations from the 13.28-Hz average; all other LLFM2 spectra show peaks within 1 or 2 standard deviations of this average. The spectrum where this occurs is the same as that shown in Fig. 4. Figure 6 shows a temporary change in phase behavior, with the slope of phase versus frequency around 13 Hz seen to be a sensitive indicator.

There is some quantitative understanding of phase behavior near 13 Hz, based on a simple model of two mechanical motions at slightly different frequencies (Thie et al., 1986). (The specific structures in motion have yet to be identified.) The model assumes resonant transfer functions \( n_1 \) and \( n_2 \), being excited by a common forcing function \( S' \):

\[
\begin{align*}
n_1 &= (a_{11}H_1 + a_{12}H_2)S + S' \\
n_2 &= (a_{21}H_1 + a_{22}H_2)S + S'
\end{align*}
\]

(1)

Here the \( a_{ij} \) are coupling coefficients for the two motions and the two neutron detector signals \( n_1 \). Another forcing function \( S' \), such as reactivity noise, is also provided for. This model successfully computes the type of phase behavior shown in Fig. 6.

Some additional explanation of this double resonance is warranted. Even though seen in a fast reactor, the phenomenon is generic and is quite possible in PWRs. Normal behavior here is LLFM2 "seeing" a frequency 0.1 Hz higher than LLFM1. Such interference causes coherence sinks and rapid rate of changes of phase with frequency due to the close spacing, as well as PSD peaks differing for the two detectors. These three phenomena are clues to the analyst that a resonant model, such as the one here, is applicable. For the FTF this double resonance is normal; what is abnormal is a shift of this motion observation by these detectors for about 0.5 h, during which both detectors "saw" the same resonant frequency and hence have little rate of change of phase near 13 Hz.

5. DISCUSSION

In this trial use of database management software for noise analysis assistance, some encouraging features become apparent:

1. Any specified feature of functions commonly used in noise analysis may be examined in all data taken, rather than just spot-checked as would be the case without this proposed tool.
2. With large numbers of answers for specific quantifiers available for averaging, better accuracies are achievable and statistics of answers can be studied to ascertain whether rare events are statistically significant.
3. Screenings of data for the presence or absence of highly specific cases obeying selected--even complex--criteria are quite easy. Programs involving sequences of simple commands are capable of automating the entire process for continued on-line surveillance if desired.
4. This method somewhat stimulates the analyst to formulate logical inquires into the data; this supplements passive visual searches for the unusual.
5. Data qualification by comparison to any set of rules one might formulate is a straightforward application.
6. Varities of display (see Figs. 3 and 5) and of intercomparisons are possible and are limited only by the analyst's imagination.
7. Unlike pattern recognition and some other automated data-handling methods, an analyst's interactive approach to the data's features is a natural way to use this tool (i.e., one answer somewhat suggests the next question). Intimate contact seems to be preserved in spite of an intimidating volume of data.

Finally, a few advisory comments should be made regarding further use of the approach proposed here:

1. User-friendly programs with numerous and powerful commands should be used to avoid the distractions of programming intricacies and to minimize obstacles to trying new approaches.
2. The approach of compressing data into specific features is encouraged because of computing time considerations, though this becomes less significant as computers improve. However, one should be alert to the possibility of more information existing in the individual data points and should not always restrict analyses to just a few features.
Fig. 6a. Phases between LLFM1 and LLFM2 during a temporary anomaly on January 12, 1987. The slope in deg/Hz is not excessively large in the 13-Hz region.

Fig. 6b. Phases between LLFM1 and LLFM2 a 0.5 h after the January 12, 1987 anomaly, as representative of normal behavior. With these detectors sensing interfering resonances 0.1 Hz apart, there is an excessively large slope near 13 Hz.

3. Close contact with the data by visual examination of samples should be maintained. Automated manipulations should not be allowed to insulate the analyst from the data.

6. CONCLUSIONS
Using database management software for extracting and handling features from large volumes of noise data functions, such as spectra, has been explored and found useful. Anomalies that might otherwise be buried can be discovered. In application to FPTF data, it was found that some additional insights into previously studied phenomena were gained. The example of an interfering double resonance here can alert analysts to the possible occurrence of similar phenomena in other reactors.
7. ACKNOWLEDGEMENT

The author is indebted to B. Damiano of the Oak Ridge National Laboratory for his efforts in overseeing the data collection system, and to L. R. Campbell and the FFTF staff for providing information about the test.

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Microrim, Inc. (1986) R Base System V, Redmond, WA.
A REAL-TIME BWR STABILITY MEASUREMENT SYSTEM

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Abstract - This paper describes the characteristics of a portable, real-time system used for nonperturbational measurements of stability in boiling water reactors. The algorithm used in this system estimates the closed-loop asymptotic decay ratio using only the naturally occurring neutron noise and it is based on the univariate autoregressive methodology.

INTRODUCTION

The main purpose of this paper is to present our experience in the development of a real-time boiling water reactor (BWR) stability measurement system. This portable system is the culmination of several years of research in the BWR stability field, and it has been applied already in two series of stability tests. In this paper, the technical foundations for BWR stability measurements using noise analysis are presented first, followed by some details of the implementation of the stability monitor. The results of two tests using this monitor are given, along with results from a previous set of stability tests under single-loop operating conditions.

The present real-time measurement system is not intended to be used for continuous stability monitoring in a reactor. Its main purpose is to perform on-demand tests whenever a confirmation of the reactor stability is needed to satisfy either regulatory or operational concerns. Noise-based stability measurement systems are not well suited for continuous monitoring for two reasons: Reactor stability is a function mostly of the void-fraction distribution in the core, and it only becomes relevant for safe operation at very high power to flow ratios (for instance, at natural circulation) that typically correspond to unusual nonsteady state operating conditions. Second, these stability measurement systems yield stochastic results, which need to be averaged over a period of 10 to 30 minutes to obtain reasonable accuracy. Thus, by the time a continuous stability monitoring system alerts the operator that he is approaching an unstable condition, the reactor has probably scammed. There is a need, though, for on-demand tests to confirm the reactor stability under operating conditions that are not well understood. For instance, stability tests might be required to evaluate the effect of new fuel types, to determine the reactor's stability under unusual conditions, or to confirm the results of computer calculations.

Within the above context, our objective was to develop a portable system that would produce, in an automatic mode, an on-line estimate of the reactor stability.

**BWR STABILITY BACKGROUND**

Under normal operating conditions, BWRs are absolutely stable and controllable. Only high-power-density BWRs are susceptible to instabilities and only when operated at power-to-flow ratios that result in high void fractions. Nevertheless, the possibility of experiencing these instabilities is real, and they have been observed in at least three commercial BWRs (Waaranpera and Anderson, 1981; Sandow and Chen, 1983; Gialdi and colleagues, 1984). For this reason, United States utilities are required by the Nuclear Regulatory Commission to evaluate the reactor stability for every reload core under plant technical specifications provide for monitoring of neutron flux oscillations in the so-called limit-cycle "Detect and Suppress" (D&S) region. This region is defined by these specifications and commonly lies below the 40% flow line and above the 80% rod control line. Within this region, the reactor operator must monitor average and local power oscillations to detect instabilities. Should instabilities occur, they should be suppressed either by inserting control rods or by increasing recirculation pump speed.

Stability tests have been performed for calculation benchmark purposes or for verification of reactor stability under special operating conditions (March-Leuba and colleagues, 1984 and 1986), and new tests are planned. When an instability was reached in previous tests, the reactor safety was not compromised, and calculations showed that fuel integrity would be maintained even if the power oscillations were more severe than those observed. However, a real-time estimate of the reactor stability during such tests is desirable to avoid unforeseen problems. In this manner, reactor operation in the D&S region can be accomplished safely, because a real-time stability estimate can warn of the approach to an unstable condition before limit cycle power oscillations are established.

The underlying cause for the observed BWR instabilities is the coolant density reactivity feedback. This is a negative feedback, and, as such, the reactor is normally stable. With high core void fractions, though, this feedback becomes so strong that it induces oscillations at a frequency of -0.5 Hz. As the strength of feedback increases, the oscillations become more pronounced and an oscillatory type of instability can be reached. In the frequency domain, these oscillations correspond to a resonance, which is obvious in the reactivity-to-power transfer function at a frequency of -0.5 Hz (Fig. 1).

The parameter of merit for stability calculations or measurements is the asymptotic decay ratio (DR), which is defined as the limit of a series formed by the ratios between consecutive maxima in the impulse response of the dynamic system (March-Leuba and Smith, 1985). The asymptotic DR is directly related to the position of the most unstable pair of complex poles, and it is a well-defined quantity in the sense that the system is guaranteed to be stable as long as the DR is less than 1.0. This is not the case with the apparent DR (defined as the ratio between the second and first peaks) that is not directly related to the system's stability (March-Leuba and Smith, 1985).

**RELATIONSHIP BETWEEN BWR NOISE AND STABILITY**

Experience has shown that BWR neutron noise has two main components: The first is dominant at frequencies higher than 1 Hz and its main characteristic is that it is axially correlated but radially uncorrelated. The second component, which is dominant at low frequencies, is correlated both axially and radially; moreover, it oscillates in phase all over the core and is correlated with process variables such as core flow and pressure. The main characteristic of the correlated component of BWR noise is a resonance at -0.5 Hz, which has been shown to correspond to the resonance in the reactor transfer function (March-Leuba, 1984).

It is believed that the major source of BWR noise is the formation, collapse, and transport of steam voids in the reactor core. The voids modify neutron absorption and thermalization, thereby introducing perturbations in cross
Fig. 1. The reactivity-to-power transfer function exhibits the characteristic 0.5-Hz resonance in a Bode Plot (a), which corresponds to a pair of complex poles in the root loci (b).
sections and, thus, in the neutron density as seen by the in-core fission detectors. Mathematically, this can be written as

\[ n(\omega) = D(\omega) G(\omega) \rho(\omega) + v(\omega) \]

where \( n(\omega) = \) neutron noise, \( D(\omega) = \) detector field of view, \( G(\omega) = \) reactor transfer function, \( \rho(\omega) = \) reactivity source term, and \( v(\omega) = \) additive neutron noise not due to reactivity changes. In terms of the above interpretation, \( v(\omega) \) represents the uncorrelated part of the neutron noise, while the first term of Eq. (1) represents the correlated part.

The reactor's stability is determined by the most unstable pair of conjugate poles of \( G(\omega) \). Thus, assuming Eq. (1) holds, the neutron noise contains information about the reactor's stability because the poles of \( G(\omega) \) are a subset of the poles of \( n(\omega) \). This can be seen better with a numerical example: Assume that \( G(\omega) = 1 / (-\omega^2 + a j \omega + b) \), \( D(\omega) = 1 / (-\omega^2 + c j \omega + d) \), and \( v(\omega) = 1 / (-\omega^2 + e j \omega + f) \). Using Eq. (1), the neutron noise becomes

\[ n(\omega) = \frac{P(\omega)}{(-\omega^2 + a j \omega + b) (-\omega^2 + c j \omega + d) (-\omega^2 + e j \omega + f)} \]

where \( P(\omega) \) is a polynomial in \( j \omega \). Equation (2) states the mathematical basis for stability measurements using neutron noise. This equation tells us that the poles of the reactor transfer function are part of the poles of the neutron noise (except in the unlikely event that the zeros of \( P(\omega) \) cancel any of these poles); therefore, the stability of \( G(\omega) \) can be guaranteed if \( n(\omega) \) is stable.

Unfortunately, the poles of the reactivity source \( [\rho(\omega)] \), the uncorrelated noise \( [v(\omega)] \), and the detector field of view \( [D(\omega)] \) are also part of \( n(\omega) \); thus if an instability is detected in \( n(\omega) \), it could be due to oscillations in the reactivity source and not have anything to do with the reactor transfer function. For this reason, this type of measurement is sometimes referred to as the "output" stability, because it corresponds to the output of a dynamical system under particular excitation conditions and not to the system itself. In all cases, though, the "output" stability of \( n(\omega) \) estimates conservatively the stability of \( G(\omega) \).

**ALGORITHM TO MEASURE BWR STABILITY**

In the previous section, we have justified why the stability of a BWR can be estimated simply by analyzing the output noise. In this section we address the problem of how to accomplish this.

Equation 1 is an overly simplistic view of the world of BWR dynamics. BWRs are very complex "dynamic machines" that require extensive mathematical models to represent them; for instance, Fig. 2 shows a more realistic dynamic model of a BWR with multiple noise sources. Nevertheless, without regard to the model complexity, the basic principles of the previous section apply, and it can be shown that an equivalent mathematical model exists that contains all the dynamic information of the complete system, in the sense that both this model and the system have the same poles. This model is the equivalent of Eq. (2) in the previous section.

A technique commonly used to generate this equivalent mathematical model is the univariate autoregressive (AR) model. When properly applied, AR models extract all the dynamic information from the signal in terms of model parameters and leave a residual white noise. Then, following the arguments presented in the previous section, the stability of the AR model corresponds to the system's "output" stability under the current driving sources. This "output" stability represents, in a conservative fashion, the stability of the reactor transfer function.

Several methods have been developed to obtain the asymptotic DR from the AR parameters. The most straightforward method is to generate the AR impulse response and to measure the DR directly, while another method consists of a pole search in the Fourier transform of the AR model. All methods work well under ideal conditions, but fail in specific situations. The approach taken for this implementation was described by March-Leuba and Smith (1985) and consists of
estimating the asymptotic DR in three different ways; then, an "optimal" estimate is chosen among the three based on heuristic rules.

The algorithm to estimate the asymptotic DR from neutron noise measurements can be summarized in the following steps:

1. Compute the autocorrelation function of the average power range monitor (APRM) signal.
2. Estimate the apparent DR of the autocorrelation function.
3. Compute the AR model of "optimal" order.
4. Estimate the asymptotic DR from:
   (a) AR model impulse response.
   (b) AR model frequency-domain pole search.
5. Validate the DR estimates from steps 2, 4a, and 4b using heuristic rules (e.g., the oscillation frequency must be between 0.3 and 0.8 Hz).
6. Select the largest valid DR as the result. This is a conservative estimate of the reactor's asymptotic DR.

THE REAL-TIME IMPLEMENTATION

The above algorithm has been implemented in the portable personal computer shown in Fig. 3. The data acquisition hardware is a standard, commercially available two-channel analyzer with computer controlled filters and amplifiers.

The real-time characteristics of the present implementation of the algorithm are achieved by computing short-term correlations (approximately one every 1.5 minutes) that are pooled to update the average correlation function. Current- and average-DRs are obtained from the current and average correlations, respectively. When enough blocks have been sampled, the average DR should converge to a value representative of the reactor stability. Unless the DR is close to 1.0, the current-DR estimate always has poor statistical precision because the amount of time needed to adequately determine the DR is inversely proportional to the reactor's stability. The motivation for providing a real-time current-DR estimate is to alert the operator of the approach to an unstable condition during power maneuvering between test points.

The output of the present stability measurement system is mainly graphical. Figure 4 presents a typical screen dump, which contains the current and average
Fig. 3. Portable real-time stability measurement system.

Fig. 4. Typical screen at the end of a stability analysis run.
Table 1. Results of Susquehanna-2 and Grand Gulf-1 tests.

<table>
<thead>
<tr>
<th>Test point</th>
<th>Power (MWh)</th>
<th>Flow (Mlb/hr)</th>
<th>Decay ratio</th>
<th>Frequency (Hz)</th>
</tr>
</thead>
<tbody>
<tr>
<td>SUSTLO</td>
<td>1970</td>
<td>46.8</td>
<td>0.33</td>
<td>0.39</td>
</tr>
<tr>
<td>SUSSLO</td>
<td>1834</td>
<td>43.9</td>
<td>0.37</td>
<td>0.34</td>
</tr>
<tr>
<td>GGTB1</td>
<td>1997</td>
<td>44.4</td>
<td>0.31</td>
<td>0.44</td>
</tr>
<tr>
<td>GGTB2</td>
<td>2363</td>
<td>50.5</td>
<td>0.32</td>
<td>0.44</td>
</tr>
<tr>
<td>GGTB4</td>
<td>2257</td>
<td>44.3</td>
<td>0.32</td>
<td>0.43</td>
</tr>
<tr>
<td>GGTB6</td>
<td>1698</td>
<td>29.8</td>
<td>0.35</td>
<td>0.37</td>
</tr>
<tr>
<td>GGTB8</td>
<td>1745</td>
<td>33.6</td>
<td>0.37</td>
<td>0.40</td>
</tr>
</tbody>
</table>

correlations along with the average power spectral density (PSD) and the estimated DRs. A trend showing the convergence of the average-DR is also displayed.

Since these DR estimates are based on a noise measurement, the results are stochastic in nature and need to be averaged over a length of time. It is important to emphasize the stochastic nature of the results by including an error band for the estimates. Based on our studies, we have found that one of the major sources of error in stability measurements is the determination of the "optimal" model order. For this implementation, we have used the well-known Akaike's information criterion (AIC) to determine the optimal order. The result displayed, though, is an average of the DRs for five model orders (two above and two below AIC's optimal order). This allows averaging process for the calculation of an error estimate, which is obtained from the dispersion of the five DR estimates. In addition, the statistical error is taken into account by measuring the dispersion of the past current-DR estimates.

APPLICATION TO STABILITY TESTS

The present system has already been applied in two series of tests to evaluate the stability of commercial BWRs (March-Leuba and Fry, 1987). The Susquehanna-2 tests were conducted in November 1986, and the Grand Gulf-1 tests were performed in January 1987. The purpose of both tests was to evaluate the stability of the new reload cores in the respective reactors. The results of the application of the stability measurement system were very satisfactory. In both cases, it was shown beyond reasonable doubt that the reactors were very stable. Table 1 contains a summary of the operating conditions, measured DRs, and oscillation frequencies.

SINGLE-LOOP OPERATION

On February 1985, a series of tests were conducted in the Browns Ferry-1 reactor to determine the source of a significant increase (on the order of 300%) in noise magnitude observed while operating with only a recirculation loop active (March-Leuba and colleagues, 1986). In particular, it was of interest to determine whether this increase in noise was due to a possible core instability.

Single-loop operation (SLO) is not a normal condition of operation for BWRs. While in SLO, the inactive loop becomes a parallel path to the core for the water being pumped by the active loop. When the active loop flow is increased, the lower plenum pressure can overcome the static pressure head of the inactive jet pumps and, thus, establish reverse flow. Reverse flow through the inactive pumps not only reduces flow through the core but causes turbulent cross flow in the downcomer region and fluctuations in the vessel water level. The observed increase in noise levels was postulated to result either from an increase in flow noise level (due to the enlarged downcomer turbulence) or to an instability (due to changes in thermohydraulic characteristics of the reactor vessel).

Analysis of the test data showed the reactor to be stable under SLO conditions and the observed noise increase to be due to increased turbulence in the downcomer region. Table 2 presents the test point conditions along with the main results of this analysis: the DR and oscillation frequency. Test points BFTB0
Table 2. Results of Browns Ferry Single Loop tests

<table>
<thead>
<tr>
<th>Test point</th>
<th>Power (MWe)</th>
<th>Flow (Mib/hr)</th>
<th>Decay ratio</th>
<th>Frequency (Hz)</th>
</tr>
</thead>
<tbody>
<tr>
<td>BFTP0</td>
<td>3185</td>
<td>104.0</td>
<td>0.29</td>
<td>0.75</td>
</tr>
<tr>
<td>BFTP1</td>
<td>2174</td>
<td>57.4</td>
<td>0.34</td>
<td>0.47</td>
</tr>
<tr>
<td>BFTP2</td>
<td>1769</td>
<td>38.9</td>
<td>0.45</td>
<td>0.41</td>
</tr>
<tr>
<td>BFTP3</td>
<td>1538</td>
<td>32.8</td>
<td>0.53</td>
<td>0.40</td>
</tr>
<tr>
<td>BFTP4</td>
<td>1792</td>
<td>46.2</td>
<td>0.39</td>
<td>0.47</td>
</tr>
<tr>
<td>BFTP6</td>
<td>1930</td>
<td>53.3</td>
<td>0.34</td>
<td>0.46</td>
</tr>
<tr>
<td>BFTP7</td>
<td>1779</td>
<td>53.3</td>
<td>0.38</td>
<td>0.46</td>
</tr>
</tbody>
</table>

through BFTP3 represent two-loop operation (TLO), while tests BFTP4 through BFTP6 were conducted under SLO conditions. It can be observed in Table 2 that, stability-wise, there is no significant difference between TLO and SLO. Therefore, by applying the noise analysis stability measurement algorithm, we were able to rule out instabilities as the cause of the noise level increases.

The main characteristics of the noise during both SLO and TLO are presented in Figs. 5 (time domain) and 6 (frequency domain). These figures indicate that the noise increased over a wide frequency range and affected all process variables. By means of a transfer function and a coherence analysis, we were able to identify the source of the observed noise increase as an increase in turbulence or as possible water level oscillations in the downcomer region. Indeed, when the flow-to-power transfer functions for SLO and TLO are compared for similar power and flow conditions, they show a remarkable agreement even though the magnitude of the noise is 300% higher in the SLO case (Fig. 7). These measured

![LOOP A FLOW (11-20)]

![LOOP B FLOW (11-10)]

![JET PUMP 11 DP]

![JET PUMP 10 DP]

![CORE PLATE DP]

![REACTOR PRESSURE]

![PUMP A DISCHARGE FLOW]

![APRM B]

![LPRM 16-41 B]

![LPRM 24-17 B]

![LPRM 32-33 C]

![LPRM 48-25 B]

![TIME (s)]

(a) BFTP1

(b) BFTP6

![LOOP A FLOW (11-20)]

![LOOP B FLOW (11-10)]

![JET PUMP 11 DP]

![JET PUMP 10 DP]

![CORE PLATE DP]

![REACTOR PRESSURE]

![PUMP A DISCHARGE FLOW]

![APRM B]

![LPRM 16-41 B]

![LPRM 24-17 B]

![LPRM 32-33 C]

![LPRM 48-25 B]

![TIME (s)]

Fig. 5. Time traces of reactor signals (a) test BFTP1 (TLO) and (b) test BFTP6 (SLO).
Fig. 6. Comparison of normalized power spectral densities for test BFTP1 (TLO) versus test BFTP6 (SLO). (a) Neutron noise and (b) flow noise.

Fig. 7. Comparison of transfer functions from core-plate pressure drop to average power range monitor in TLO and SLO with similar power and flow conditions. (a) Tests BFTP1 vs BFTP6 and (b) Tests BFTP2 vs BFTP4.
transfer functions also have the shape expected from theoretical analysis (e.g., compare Fig. 1 with Fig. 7). Note, though, that the measured transfer functions are open-loop; thus, they cannot be used to obtain stability estimates, because the open-loop flow-to-power transfer function lacks the recirculation loop dynamics. Experimental evidence has shown that open-loop DRs can be as much as 50% smaller than closed-loop DRs (March-Leuba and Otaduy, 1985), and thus, they are nonconservative estimates of the reactor stability.

**SUMMARY**

This paper presents our experiences related to the development and applications of a real-time BWR stability measurement system that is based on a noise analysis technique. The results of three series of stability tests in which this technique was used have also been discussed. Overall, we are very satisfied with the performance of the present system and expect that it will probably be used again to evaluate the stability of commercial BWRs when there are doubts regarding the accuracy of computer calculations due to abnormal operating conditions, such as single-loop operation or new fuel types.

**REFERENCES**


SYSTEMS (PART II)

Session chairman: N. Suda (Japan)
SUMMARY OF THE SESSION

This session is the second one in the series of sessions on systems for reactor noise analysis, and thus is more or less the continuation of the preceding one. Emphasized are the pattern recognition (Saedtler, Bokor and Kemeny) and the artificial intelligence expert systems in particular (Dach, Kitamura, Bokor and Kemeny). The main objectives of study are sophisticated selection and combination of various branches of signal processing technology (Kitamura and Bokor), and development of intelligent systems for vibration monitoring and diagnostics (Saedtler and Dach). From the computer application point of view, the symbolic of logical manipulation, in addition to ordinary numerical computation, plays a key role in most of the studies.

Saedtler described a diagnostic system based on the combination of feature extraction and classification. The feature vector is composed of specific functions of auto power spectral density values at discrete frequencies. The classification is basically a hypothesis test and, assuming that the feature vector is related to a stochastic process described by a Markov model, a maximum likelihood model, called the maximum-a-posteriori (MAP) estimator, is obtained. The hardware is a modular system composed of 16 bit-type microprocessors. He shows a sample plot of a simulated decision process.

Dach discussed the development of an expert system for diagnostics of reactor internal behaviour. The knowledge concerning measured noises, such as deviation of frequency/amplitude of fluctuation power, is augmented by knowledge derived from a mathematical model of the mechanical behaviour of a reactor. The knowledge base is a set of production rules thus obtained. In order to identify the anomaly origin, use is made of decision matrices, which relate increase/decrease of the noise descriptor values to transient in reactor operation regime.

Kitamura described a database for supporting the reactor noise analysis. In order to extract appropriate information from the reactor noise, the expertise in the signal processing technology is needed in every stage of analysis, namely, a priori tests of statistical properties of noise, modelling and noise signature evaluation, and a priori tests of the results. The frame representation of knowledge specific to procedures of analysis is combined with a set of production rules related to the situation encountered in the analysis. He shows several examples which demonstrate the usefulness of the proposed method.

Bokor reviewed the functional structure of a noise diagnostic system, which is composed of subsystems for signal processing, pattern recognition, knowledge-based signal processing supervisor, diagnostic expert system, and so forth. He then described in more detail the signal processing supervisor, the objective of which is similar to that of the system developed by Kitamura.

Kemeny extended his view on pattern recognition and artificial intelligence and mentioned some of his experiences.
A MODULAR MULTI-MICROCOMPUTER SYSTEM FOR ON-LINE VIBRATION DIAGNOSTICS

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Federal Republic of Germany

ABSTRACT

A new modular multi-microprocessor system for on-line vibration monitoring and diagnostics of PWRs is described. The aim of the system is to make feasible an early detection of increasing failures in relevant regions of a reactor plant, to verify the mechanical integrity of the investigated components, and to improve therefore the operational safety of the plant.

After a discussion of the implemented surveillance methods and algorithms, which are based on hierarchical structured identification (estimation) and statistical pattern recognition tools, the system architecture (software and hardware) is portrayed. The classification scheme itself works sequential so that samples (or features) come arrive on-line. This on-line classification is important in order to take necessary actions in time. Furthermore, the system has learning capabilities, which means it is adaptable to different, varying states and plant conditions.

The main features of the system are presented and its contribution to an automation of complex surveillance and monitoring tasks is shown.

1. INTRODUCTION

With the new generation of VLSI components, microprocessors and the proper tools it is now possible to implement more and more complex procedures and algorithms into small and inexpensive systems. For example, such equipment are surveillance and diagnosis systems for nuclear reactor plants. The methods used by the computer codes are described by the following keywords: Comprehensive signal preprocessing, filtering, identification and estimation, classification (pattern recognition), and - in a development stage - expert systems. In particular, diagnosis techniques can be subdivided into the approaches

- statistics (FFT, correlation, decision-making, ...),
- auto-regressive modelling,
- Kalman filtering and estimation,
- fuzzy set theory and
- artificial intelligence (knowledge engineering).

The aims of the diagnosis systems are, to improve the safety of the plants, and to increase the maintainability and availability.

For nuclear power plants (especially for PWRs) one point of interest is the surveillance of the vibration behaviour of the pressure vessel, its internals, and other primary system components. Meanwhile, a well-founded knowledge about these system characteristics has been accumulated so that automation is an important consideration for efficient operation of monitoring systems. In this paper a modular microprocessor based system for this task is described. The system runs a real-time operating system and the application algorithms are based on identification and pattern recognition techniques.
2. SURVEILLANCE AND CLASSIFICATION ALGORITHMS

The vibration signals, derived from the different sensors installed at the reactor plant, are of stochastic nature containing deterministic parts. In general, there are four types of sensors: Absolute and relative displacement, pressure, and neutron flux sensors. Short time histories of the sensor signals are shown in figure 1.

![Sensor Signals](image)

**Fig.1:** Time histories of the sensor signals

Since the signals are quasi-stationary, they can be processed using commonly known fast Fourier transform (FFT) techniques to obtain the power spectral densities (PSDs).

### 2.1 Feature extraction

In a pattern recognition terminology, the computation of the spectral densities is a feature extraction process. Thus the feature vector is composed of the spectral density values at the selected discrete frequency points respectively of a measure in a predefined frequency band. An estimate of the auto power spectral density (APSD), applying the periodogram method, is given by

$$\hat{S}(f_1, t) = \frac{1}{J} \sum_{j=1}^{J} \frac{|Y_j(f_1, t)|^2}{T}$$  \hspace{1cm} (1)

where

- $\hat{S}(f_1, t)$ = auto power spectral density vs. frequency and time
- $Y_j(f_1, t)$ = Fourier transform of the quasi-stationary (windowed) signal $y(t)$ in subsection $j$,
- $T$ = length of subsection (seconds) used to estimate the Fourier transform with a FFT algorithm,
- $J$ = number of subsections,
- $f_1$ = discrete frequency, $l=1, \ldots, L$,
- $t$ = time, absolute scaled, at which the FFT is calculated,
- $L$ = number of discrete frequencies,
- $|.|$ = absolute value.

With the above considerations and assumptions the feature vector of the classification or surveillance problem could be defined as

$$z(t) = [g_1(\hat{S}(f_1, t)), \ldots, g_N(\hat{S}(f_j, t))]$$  \hspace{1cm} (2)

where
A modular multi-microcomputer system for on-line vibration diagnostics

\[ g_i(\hat{S}(f_i,t)) = \text{function of APSD values at discrete frequencies with} \]
\[ i = 1, \ldots, N, \]
\[ N = \text{dimension of feature vector (number of features)} \]
\[ \{.\}' = \text{transposed.} \]

The functions \( g_i(.) \) can be chosen as

\[ g_i(.) = \sum_k \log \hat{S}(f_k,t) \]

or

\[ g_i(.) = \sum_k \{ \log \hat{S}(f_k,t) \cdot [f_k-f_m]^e \} / \sum_k \log \hat{S}(f_k,t) \]

with

\[ \sum_k = \text{sum over a frequency interval of length } K \text{ (inclusive } K=1 \text{ for the first definition),} \]
\[ f_m = \text{middle frequency of the interval,} \]
\[ e = 1, 2, \ldots \text{ (moment order).} \]

The vector \( z(t) \) is also of stochastic nature, since random plant and measurement conditions exist from measurement period to measurement period. Thus, we have to implement statistical observation classification methods, as discussed in the sequel.

\( N \) is of order 10 for one APSD, depending on the number of peaks in an APSD which are necessary for diagnostic purposes. But, the number of components of the feature vector increases drastically if several APSDs, calculated for different signals, are concatenated. In addition, the components of \( z(t) \) are, in general, not statistically independent. This is especially the case for neighbouring spectral lines. To overcome these problems, suitable decomposition algorithms are introduced and an accommodated modular parallel computer architecture is designed.

In general, a statistical pattern recognition system consists of three fundamental processing blocks as shown in figure 2.

![Fundamental structure of a numerical pattern recognition system](image)

**Fig.2:** Fundamental structure of a numerical pattern recognition system

The aim of this configuration is to optimize the entire recognition problem in such a way that a separation of the pattern, belonging to different classes, can be obtained with a minimum of risk or cost in the Bayesian sense. A truly optimal solution of the recognition system that optimizes jointly the preprocessing, feature selection and extraction, and classification does not exist, except for very simple cases. Therefore, the problem is solved more heuristically or with experienced "engineering judgement". E.g. feature selection and extraction is done using the relations given by Eq. (1) to (3). These features are very informative for stochastic excited and disturbed mechanical structures and components.

2.2 **Classification**

The feature vector, defined in the last section, is an input to the classifier block (Fig.2). Thus, for our problem we have to detect any anomalous behaviour of the features. If such an effect can be attributed to an irregular operational status of the investigated component a message should be given to the system operator. To solve this detection problem a strategy is chosen which is known in decision theory as a hypothesis test.
Because of the on-line and real-time requirements the algorithms must work sequentially in time. Therefore, a test was chosen which has this properties. It is known in the literature as Wald's sequential probability ratio test (e.g. Sage, 1971). Several extensions and modifications of this test are published. A simplified version of Wald's sequential probability ratio test can be written in the following form:

Decide for hypothesis (or class) $H_{ki}$ at time $t$ if

$$LR_{ki} \{ Z_i(t) \} = \max_m \{ LR_{mi} \{ Z_i(t) \} \}$$ \hspace{1cm} (4)

for all $k \neq m, k > 0$,

where

$$LR_{ki} \{ Z_i(t) \} = \frac{P_i \{ Z_i(t) | H_{ki} \}}{P_i \{ Z_i(t) | H_{0i} \}}$$ \hspace{1cm} (5)

$k = 1, \ldots, M-1$

$Z_i(t) = \{ z_i(t'), t_0 \leq t' \leq t \}$,

$M =$ number of hypotheses,

$i =$ feature component,

and

$$\dot{x}_i(t) = f_i \{ x_i(t), z_i(t), t \} + G_i(t)w_i(t)$$

$$z_i(t) = h_i \{ x_i(t), t \} + v_i(t).$$ \hspace{1cm} (6)

LR is the likelihood ratio and $Z_i$ defines the feature component $i$ over a time interval. It is further assumed that each feature vector component $z_i$ is related to a stochastic process $x_i$ through the Markov model given in Eq. (6). The processes $v_i$ and $w_i$ are zero-mean white Gaussian noise terms with appropriate covariance matrices. The interconnection between the subsystems, Eq. (6), is represented by the vector $x_i(t)$, which is a function of $x_j(t), j \neq i$. This formulates the statistical dependency of the feature vector components and is motivated by the decomposition assumption of the overall system into small subsystems. If the subsystems are not connected, which can be supposed to be true in practical cases, the problem is drastically simplified, since for each feature component the decision test can be solved independently from all others. To be more general, in the following we take into account the primary form (interconnected feature components) of Eq. (6).

For our problem the hypotheses can be defined as

$$H_{0i}: z_i(t) = v_i(t)$$

$$H_{mi}: z_i(t) = h_{mi} \{ x_i(t), t \} + v_i(t),$$ \hspace{1cm} (7)

$m = 1, \ldots, M-1$.

The objective is to estimate the likelihood ratios sequentially in time for each subsystem subject to the differential equality constraints given in Eq. (6).

One possible estimate of the composed original state vector $x(t)$ throughout the interval $[t, t']$ is known as the maximum-a-posteriori (MAP) estimator. The MAP estimator is derived from maximizing the conditional probability density function $p(X(t) | Z(t))$ with respect to $X(t)$. The definition of $X(t)$ is similar to that of $Z(t)$. Now hierarchical structuring of the original large-scale system offers an effective approach to the estimation of the subsystems and thus of the likelihood ratios. A first order approximation of the recursive MAP filter algorithms with hierarchical structures was derived and coded. The necessary model and goal coordination variables are calculated in a sequential form with a predictor-corrector algorithm. A block diagram of the MAP filter is shown in figure 3. $K_i$ is the gain matrix, $P_i$ the variance matrix of the estimated state vector $\hat{x}_i$, and $G_{i}$ a vector function of the goal coordination variable.
Fig. 3: Block structure of the MAP filter

An estimate of the likelihood ratio (natural logarithm) is then given as

$$\ln[LR_{mi}(Z_i(t'))] = -\frac{1}{2} \int_{t_0}^{t'} \Gamma_i dt + S_{mi}$$  \hspace{1cm} (8)

where $\Gamma_i$ and $S_{mi}$ are functions of the feature components and the estimates derived from the MAP filter equations. $S_{mi}$ is a sufficient statistic. Using different initial conditions for the MAP filter, especially variance values of the state vectors, it is possible to adapt the estimates more or less to the feature components. Thus, the learning capabilities of the pattern recognition system are introduced by the filter algorithm initializations. A detailed description of the algorithms is given by Saedtler (1982, 1984, 1985).

2.3 Model of feature components

After a description of the mathematical framework of the problem we are now able to formulate the model equations for our specific problem.

Recalling the definition of the feature vector $z(t)$ in Eq. (3) and taking into account a special connection of the state space models $x_i(t)$, Eq. (6), a model of the subsystem $i$ is given by

$$x_i(t) = a_i + b_i \pi_{i-1}(t) + c_i \pi_{i+1}(t) + w_i(t)$$
$$z_i(t) = x_i(t) + v_i(t)$$  \hspace{1cm} (9)

The parameters $a$, $b$ and $c$ are fixed after an initial identification or learning step and can be - if necessary - adapted to the actual status of the plant from time step to time step. To estimate the unknown parameters, the state vector $x_i(t)$ is augmented by "random-walk" models for these parameters.

The model coordination variable $\pi$ in Eq. (9) represents the interconnection conditions between the feature components. Only a correlation with the neighbouring components is assumed. Practically, the model describes the time behaviour of each feature vector component by a straight line with initially unknown slope and coupling coefficients with its neighbours. A di-graph of the overall model is shown in figure 4.

As an example, consider the decision problem of four pattern classes, then a realistic assumption for the hypotheses could be

$$H_{0i}: z_i(t) = v_i(t)$$
$$H_{mi}: z_i(t) = x_i(t) + \delta_m + v_i(t) \hspace{1cm} m = 1, 2, 3$$

where

$$\delta_m = \{0.0, -0.06, +0.06\},$$

and $x_i(t)$ is normalized to about 1.0.
Solving the MAP filter equations sequentially in time and computing an estimate of the likelihood ratios we can perform the decision test as given by Eq. (4). The test strategy could be as follows: (1) set the initial values of $x_i$ for the actual test to the final values of the previous test for which the hypothesis $H_i$ was accepted, (2) set the initial variance values of $x_i$ to the original values for each test.

A performance measure of the decision test is its conditional error probability. An upper bound of this measure is the so-called rejection threshold $\lambda \leq (M-2)/(M-1)$, $M>2$. The conditional error probability for the hypothesis test is defined as

$$
E \{ Z_i(t) \} = \max_j LR_{ji} \{Z_i(t)\} / \sum_{m=1}^{M-1} LR_{mi} \{Z_i(t)\} 
$$

Fig.4: Di-graph of the overall model

3. SYSTEM ARCHITECTURE

After the introductory description of the methods and algorithms a realization of the monitoring and surveillance system should be outlined.

3.1 Hardware design

The system was designed as an "open", modular system so that more computer power could be added without extensive redesign. A sketch of the realized multi-master multi-microprocessor system is shown in figure 5.

Installed are 16 bit-type microprocessors with attached numeric co-processors. A parallel priority resolution scheme, using extra logic, allows the bus masters to acquire and control the global bus interface. The system is called SPIRIT, which is a synonym for "statistical pattern identification, recognition and interpretation tools".

One block in the figure signifies a complete single board computer (SBC), controller or memory device, respectively. The so-called "supervisor" processor, together with the controller, manages the transfer of program and data files to and from a mass storage device (floppy disk and Winchester disk) and assists as communication tool for the system user. In addition, a hardware real-time clock is mounted at the local bus of the supervisor processor. This clock/calendar module functions as time basis for the system. A terminal (CRT) is connected to the SBC via a serial interface and the parallel interface is equipped with a line printer.

Besides of the on-board RAM of each dedicated computer supplementary RAM devices are installed. These devices are connected directly to the global system bus, which is an industrial standard 16-bit bus. The two other SBC's with the analog-to-digital converters (ADC's) on its local bus unit run the application programs and communicate with the supervisor processor ("peer to peer device" communication). A simplified diagram of the ADC-module is shown in figure 6.
Fig. 5: Multi-microprocessor system

Fig. 6: Analog input module board
At this time, two analog signals are connected via a conditioning panel to each SBC. This is done because of the FFT algorithm characteristics; the algorithm requires a complex input string (two real arrays are concatenated).

An additional board on the global bus functions as a graphics output subsystem. This unit is managed through a driver implemented in the supervisor processor software. The graphics controller itself generates with the help of on-board hard- and firmware the video signals which drive a high-resolution color-display monitor. With the video printer, connected to the monitor, hard-copies of the screen can be taken manually. The software of the graphics system is based on the standard CGI (computer graphics interface) of ASNI (draft version).

3.2 Software design

The operating system for the multi-processor architecture is real-time oriented so that external events (e.g. interrupts from the ADC module) can be handled very quickly. The main capabilities of the software executive are

- object-oriented architecture,
- real-time priority-oriented scheduler,
- multi-tasking and multi-programming managements,
- supports multi-processing on the global system bus.

Thus, with these tools it is possible to handle the surveillance/diagnosis problem in real-time and to use the concept of interprocessor communication for message transfer between the supervisor SBC and the other "peer devices" or masters. Each SBC runs a dedicated configured copy of the operating system. The application programs are written in a high level language. A flow chart of the programs, loaded into the SBC's with the ADC's, is shown in figure 7.

Excepting some preparatory steps, the digitized signals are preprocessed and the features are extracted, then the likelihood ratios are estimated by use of the hierarchically structured MAP filters. After each completed processing step the results are transferred to the supervisor processor for further manipulation. The code is split up into several tasks which are synchronized by use of semaphore techniques. For the input signals a double buffering method is implemented. If one buffer is filled reading values from the ADC module, the other is processed iteratively. The data transfer to and from the supervisor processor is done in the communication job, in which the message exchange task is the active part. The messages are sent to a "shared memory" area which belongs to all processors and then the receiving processor copies them into its private memory. The memory layout of the multi-processor system is strongly linked to the software configuration process, which must be done for each agent (or device).

As development environment the target system itself was used, because the necessary tools (compiler, linker, locator, editor etc.) are available for the real-time operating system.

3.3 Example

A sample plot of a simulated decision process is shown in figure 8. This figure is a reproduction of a hard-copy taken from the graphics monitor.

The input signal for this example was a sine wave. The spectrum of the signal is shown in the diagram on the bottom of the plot. The deviations during a time period of about 3.7 hours are indicated (light background around the spectrum). After about 1.5 hours from the start of the test the frequency of the sine wave was shifted for nearly 1 Hz. The chosen feature represents the sum over the frequency band around 40 Hz stressed out in the plot. The two small diagrams in figure 8 show the "measured" and estimated feature and the conditional error probability versus time. The algorithms are initialized to learn the above mentioned deviation so that after some time steps the estimated feature (the reference feature) reaches the actual ("measured") feature asymptotically. The hypothesis test results are shown in the diagram indicated with "decision". The test period for this case was 100 seconds, and the time to estimate, extract and process the features was about 28 seconds.
Fig. 7: Flow chart of the code running on one application processor.
4. CONCLUSION

In the paper a modular vibration monitoring system for nuclear reactor components based upon pattern recognition techniques for large-scale problems has been presented. The main advantages of the introduced method are as follows,

- modular design based on a suitable separation of the feature vector,
- flexibility due to independency from the feature vector type,
- adaptability to different process states, and
- real-time processing of decision functions with sequential algorithms.

The structure of the problem makes the implementation on parallel computing facilities an attractive proposition. Thus, the design and confectioning (soft- and hardware) of a dedicated, modular multi-microprocessor based system was a consequent conclusion.

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REFERENCES


DEVELOPING A KNOWLEDGE BASE FOR NOISE DIAGNOSTIC EXPERT SYSTEM OF REACTOR INTERNAL BEHAVIOUR

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Abstract - Noise diagnostic expert system is built for determination of the origin of noise descriptors change during operation from that in stationary state. The series of IF-THEN rules is compiled for event-oriented analysis of core-barrel pendulum motion investigation. The theoretical model for core uncovering identification as a function of core barrel vibration was developed. For distinguishing the influence of flow regime and boiling noise from in-core vibration on ex-core detector behaviour, the autoregressive model is used. The fundamental relations between knowledge base and the verification of data base validity is mentioned. For the prediction and analysis of strains conditions of safety related components of WRER-440 reactor, a mathematical model was developed. Reactor pressure vessel, cover and internals are represented as a discrete vibrating system with 24 degree of freedoms.

1. INTRODUCTION

Reactor noise techniques have seen important developments and applications over the past few years. The actual state of noise analysis application could be characterized by using semiautomatic or full-automatic monitoring of noise spectral characteristics together with their archivation and subsequent off-line evaluation. The evaluation is for a great part the routine work, but in complicated cases there is a great need of close cooperation with specialists of different professions. The introducing of expert systems in the field of noise diagnostics refers namely to the very problems of evaluations, e.g. the solution of proper diagnostics problems.

2. THE BASIC PHILOSOPHY OF NOISE DIAGNOSTIC EXPERT SYSTEM

Since the beginning of 80th, the computer-aided diagnostic systems proposed for prevention and mitigation of an accident in nuclear power plants have been developed. Lately the methods of artificial intelligence were applied to identify the origin and type of abnormal operational state of the plant (Nelson, 1982; Cain, 1986; Hajek, et al., 1986, Yokobayashi, et al., 1986).

Noise diagnostics of reactor internals behaviour is connected very closely with the analysis of neutron noise signals. Neutron noise fluctuations are carrying information on dynamic behaviour of in-reactor process. The diagnostics of mechanical vibrations of reactor internals from the neutron noise is an indirect method, depending on physical transformation of mechanical motion into the fluctuations of neutron flux.

PNE-H
The abnormal vibrations aggravate the material strain and may be a source of latent damage component integrity, which could cause a shortage of component or even subsystem life expectancies (UCRL-15103, 1977).

These abnormal vibrations are the cause of damage, which could not be measured directly, but by means of signal analysis the abnormal amplitudes of vibrations or the changes of vibration modes, etc. may be identified. Successive expertise could determine the technical state or the origin, which has led to the change in technical state.

The main requirements for an expert diagnostic system are:
- monitoring of equipment state in all operation situations,
- detection of deviations from "normal" operation conditions,
- classification of phenomenon significance,
- recommendation for solving the existing situation.

In the Fig. 1, there is a scheme of simplified structure of noise diagnostic expert system, which pursue the logical process used by solving practical diagnostic problem.

![Diagram of noise diagnostic expert system]

**Fig. 1. Simplified structure of noise diagnostic expert system**

This is a synthesis of existing knowledge about specified problems in knowledge base and information about the state of diagnostic object in data base complemented by numerical simulation results.
The incorporation of noise diagnostic expert system into the system of reactor technological system control can be seen in Fig. 2.

![Diagram showing the incorporation of noise diagnostic expert system into the system of reactor technological system control.]

Fig. 2. Noise diagnostic expert system and risk assessment within the system of reactor technological process control.

Technological processes characterized by a state quantity \( \{x\} \) is regulated by a quantity of action \( \{U\} \). The check out quantity \( \{Y\} \) gives informations about technological process and the system itself. Quantities \( \{N\} \) characterize the failures (Vavfin, 1986). The proper diagnostic is realized by means of three blocks: Identification of state - failure recognition - risk assessment.

3. BUILDING OF KNOWLEDGE BASE

The efficiency of evaluation process depends among others on optimal encode of symptom characteristics and their mutual structure. The diagnostic reliability is namely a matter of completeness of knowledge base correctness of data base and validity of simulation models. We do not intend to discuss the problem of proper encoding the knowledge base into symbolic relations, but we will aim above all at the content of the structure, which recovers the noise diagnostic specification.

The application of function-oriented and event-oriented strategies could be advantageously applied. The function-oriented analysis came out of the very design of the structure using a method of response trees for solving eventual deviations. The event-oriented analysis came out of the summarization of experience behaviour of the system during operation, known accidents, series of experiments made on research reactors, non-active experimental mock-ups or computer simulations. Advantage of this strategy is quick recognition of "known" or "expected" event, but on the contrary it is unaccommodating in the case of an unexpected event.

For the efficiency of noise diagnostic expert system it is necessary to distinguish among signals faults from vendor or instrumentation channel failures and drifts. Therefore sensor data validation is considered as one integral part of expert diagnostic system. The data given to the diagnostic system go through the sensor validation routine in order to discover any malfunctioning instrument channels and ensure entrance to expert system for reliable data only. For this aim it will be inevitable to take into account the investigation of common cause failures and
instrumentation channel drifts (Nelson, 1986).

To have an effective knowledge-base, shallow knowledge must be augmented by deep knowledge of reactor structure and function. To this purpose, the permanent attention is concentrated to the development of theoretical models suitable for computerized expert system decision making. A model for solving the problem of eigenfrequencies of cylindrical shell, filled or surrounded by fluid with varying height, which is the generalized case of core uncovering was developed. Results show that a change in water level height inside the system reactor pressure vessel - core barrel depicts very significantly in the frequency spectrum of core barrel vibration and consequently in the spectrum of ex-core ionization chambres signals. This phenomenon could be also used for indication of water level in the core in the case of small LOCA or ATWS and could be a part of expert diagnostic system (Dach, Kuželka, Pečínka, 1986).

With the view to simulate mechanical behaviour of the whole primary circuit, the mathematical model, whose geometrical scheme is in Fig. 3 was developed

![Diagram](image)

Fig. 3. Geometrical scheme of primary circuit

The reactor is in this system modeled as a discrete vibrating system with 24 degrees of freedom, see Fig. 4 (Zeman, 1987). The excitation could be defined as:
- vector of frequency response for seismic calculations
- power spectral density of pressure pulsations during normal operation
- concentrated force with analysis of water impact consequence in primary circuit.
For the above mentioned cases the model is applicable within expert diagnostic system conception.

Though this model gives good survey through frequency - modal properties of all important components of the reactor and the primary system, it seems to be too complicated for own diagnostic of internal vibrations. That is why we presume to continue in investigating theoretically as well as experimentally the behaviour of core barrel as a cylindrical shell under different boundary conditions (Dach et all., 1985, 1986).

1. Pressure Vessel
2. Core Barrel
3. Lower Part of Core Barrel
4. Cover of Pressure Vessel
5. Secondary Core Barrel
BOT. Upper block
T. Support Beam
M. Distance Grids
OT. Tubes
K. Hexagonal fuel element shell
PP. Fuel rods
TT. Pressure tubes with engines
TY. Support Rod
RK. Control Rods

- discrete mass points
- flexible bond in y-direction
- flexible bonds in x and y directions
- springs

Fig. 4. Geometrical scheme of primary circuit
The fundamental strategy of expert diagnostic system based on sets of IF - THEN rules is indicated in the block diagram in Fig. 5.

Fig. 5. The fundamental strategy of expert diagnostic system
4. IDENTIFICATION OF ANOMALY ORIGIN

Only rarely we can meet with a problem to identify such an anomaly, which directly evokes an anomaly in fluctuations of noise signal. Usually the original cause is transferred to resulting anomaly by means of event sequences. There is a set of reversible and irreversible processes between cause and consequence, which creates certain structure. By logical analysis of a set of symptoms characterizing the structure a relevant syndrom could be determined and from the syndrom it is possible to derive the anomaly origin. A set of symptoms is created not only by the syndrom determined from analysis of noise signals, but also from deterministic ones.

That is why the predominant quantity of syndroms is hybrid of static and dynamic characteristics. Syndroms modelling, investigation of structure features and mutual relations between syndroms and symptoms are first-rate exercises in research in the field of diagnostic of complex objects, such as nuclear power plant. On the basis of screened models of syndroms the knowledge base of expert diagnostic system is being built.

The process of building knowledge base for identification of anomaly origin is demonstrated consecutively by using real experimental data from a nuclear power plant. The noise signals from four ex-core ionization chambers (I1-I4) and two in-core SPNDs (S1, S2) are analysed in the four following regimes:

1. Transient from nominal coolant flow in experimental fuel assembly to moderately reduced flow (approximately to 70 %),
2. Transient from the reduced coolant flow in assembly to strongly reduced flow (approximately 35 %) with oscillations (approx. ± 5 %),
3. Transient from moderately reduced coolant flow with oscillations to strongly reduced one with oscillations,
4. Transient from nominal coolant flow to moderately reduced one with oscillations.

To identify the power of fluctuations and the physical parameters, the autoregression models (Vavřín, 1986, 1987) were used. The results from the above four operation regimes are encoded into four matrices, which serve for decision making of diagnostic expert system. There are 12 columns in every single matrix corresponding to 12 descriptors (D1 - D12), chosen for each noise signal. The meaning of particular descriptors are:

D1 - Total fluctuations power, fitted by AR - model,
D2 - Partial fluctuations power corresponding to real pole of AR - model,
D3 - Partial fluctuations power of fuel assembly mechanical vibrations,
D4 - Partial fluctuations power of hydrodynamic oscillations in primary circuit,
D5 - Partial fluctuations power of coolant oscillations in fuel assembly,
D6 - Change of break-down frequency corresponding to real root of AR - model,
D7 - Eigenfrequency of fuel assembly mechanical vibrations,
D8 - Damping of fuel assembly mechanical vibrations,
D9 - Eigenfrequency of hydrodynamic oscillations in primary circuit,
D10 - Damping of hydrodynamic oscillations in primary circuit,
D11 - Eigenfrequency of coolant oscillations in fuel assembly,
D12 - Damping of coolant oscillations in fuel assembly.

In tables 1 - 4 decision matrices for described transient regimes are presented.
Table 1. Decision matrix for transient from nominal state to small reduced coolant flow without oscillations

<table>
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<tr>
<th>Detectors</th>
<th>Noise descriptors</th>
<th>D-1</th>
<th>D-2</th>
<th>D-3</th>
<th>D-4</th>
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<th>D-7</th>
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Table 2. Decision matrix for transient from small reduced coolant flow to big reduced flow with oscillations

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<th>D-2</th>
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Table 3. Decision matrix for transient from small reduced coolant flow with oscillations to big one with oscillations

<table>
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<tr>
<th>Detectors</th>
<th>Noise descriptors</th>
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<th>D-2</th>
<th>D-3</th>
<th>D-4</th>
<th>D-5</th>
<th>D-6</th>
<th>D-7</th>
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Table 4. Decision matrix for transient from nominal state to small reduced coolant flow with oscillations

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In case of no change in descriptor during evaluation of current operation regime, there is zero in pertinent place in the matrix. In case of increase or decrease of the descriptor value, there is +1 or -1 respectively.

The set of matrices creates the data base of the knowledge base. In the course of diagnostic process the current matrix which reflects the actual state of the system is determined at pre-defined time. If there are no deviations from nominal operation regime, the matrix is empty. In case of determination of some deviation, the expert system finds out from the set of decision matrices this one, which is closely related to current one. This procedure makes possible to identify the most probable event.

6. CONCLUSIONS

The applied method of anomaly identification is rough and must be understood as interim but illustrated approximation of transformation of noise analysis results into diagnostic expert system knowledge base. It is necessary to improve existing models by introducing probability and fuzzy concepts into creation of matrices and decision making process itself.

The discussed approach issued from actually applied methods of treatment of quazistationary noise signals, where only small change of statistical characteristics is expected. The application of advanced methods of nonstationary signal treatment will contribute to enlargement of noise diagnostic expert systems to the area of nonstationary transient processes.

Current activities in PSA, while they may contribute significant knowledge, are not in themselves sufficient to develop the required knowledge base for building diagnostic expert system. This is because these activities are, at present, being carried out with objectives other than the development of expert system knowledge base. An explicit program to collect, organize and integrate knowledge for inclusion in the knowledge base is required.

REFERENCES

SYNTHESIS OF HEURISTIC KNOWLEDGEBASE FOR SUPPORTING DEVELOPMENT OF GOAL-ORIENTED REACTOR NOISE ANALYSIS PROGRAMS

M. Kitamura, M. Takahashi, T. Washio and K. Sugiyama
Department of Nuclear Engineering, Tohoku University, Aoba, Sendai, 980 Japan

Abstract - A knowledgebase system which assists a noise analyst to synthesize customized, i.e. goal-oriented, computer programs is proposed in this paper. The knowledgebase stores empirical and heuristic rules for choosing a set of subroutines, selecting control parameters, evaluating validity of the analysis, and for interpreting the outcome of the computation. Various statistical test procedures were utilized extensively in order to extract information to be used as input data for the rules to make decisions. The knowledgebase system is implemented on a personal computer using PROLOG language, while the numerical analysis program is synthesized on a mainframe computer. The subroutines are combined, through consultation with the knowledgebase system, to form a specific-purpose program for given signal set and signal conditions. This approach allows us to improve efficiency of software development for noise analysis, plant monitoring and diagnosis.

1. INTRODUCTION

A wide spectrum of advanced signal processing techniques has been introduced during the last decade for information extraction from at-power reactor noise (Williams, 1977, 1982, 1985). The autoregressive (AR) modeling, autoregressive-moving average (ARMA) modeling, bispectrum analysis, cepstrum analysis, and Hilbert transformation are only a partial list of examples. However, a considerable portion of these techniques are not free from criticisms about uncertainties in the actual procedure of signal processing and in interpretation of the results. For example, the feedback system identification (Kitamura et al., 1979) and signal transmission path (STP) analysis (Oguma, 1981, 1982) led to successful results in the reported cases. In other occasions, unsuccessful or confusing results were obtained because of the difficulties in AR model order determination (Allen, 1977; Kleiss, 1983), feedback system identification (Kleiss, 1983), and in STP analysis (Por et al., 1985).

As far as our experiences are concerned, a significant portion of the uncertainties can be removed by applying appropriate expert judgements. Decisions should be made on sampling frequency, cut off frequencies of low pass and high pass filters, trend elimination procedure, window function, transformation or modeling algorithm, model order, validity of the analysis, etc. Selection of signal set is another important issue in multivariate signal analysis. Note that the standard procedures found in textbooks should be regarded only as crude guidelines. In actual practice, one has to make decisions on the basis of additional expertise. Considering the current status of computer technology for knowledge handling, it seems not only imperative but also timely to provide a tool for assisting the users in these advanced tasks.

The ultimate purpose of this work is to fulfill this need by providing a software environment for assisting development of the goal-oriented analysis programs which can execute specific tasks without bothering the analyst by the uncertainties. The development task was decomposed into three phases listed below:

(1) Collection and coordination of heuristic knowledge for signal processing and interpretation.
(2) Design and implementation of a knowledgebase system which stores the heuristic knowledge in a well-organized manner.
(3) Production of a software package for systematic development of goal-oriented noise analysis
programs.

This paper mainly describes the technical contents of phase-(1) and -(2). The phase-(3) task can be carried out without difficulty if phase-(1) and -(2) are accomplished in an appropriate manner.

The main activity at phase-(1) is introduction and evaluation of theory-based as well as empirically-derived tests for quantifying the characteristics of observed signals. Stationarity of the signal needs to be examined carefully from several different perspectives. A set of a posteriori tests are also involved, whiteness and dependency checkings of fitting residuals are examples of them. The heuristic knowledge of experts about the proper usage of various signal processing techniques is articulated in conjunction with the results of these tests. The introduction of these tests provides us with the possibility of transforming the expert's knowledge into more objective and quantitative representation. The collection and coordination of the knowledge are conducted through examination of numerical experiments specifically designed for the present purpose.

The heuristic knowledge acquired in phase-(1) was then translated into symbolic representations for synthesizing a knowledgebase system. The translated knowledge was implemented to a 16-bit personal computer by using an expert system shell developed on the basis of the PROLOG language, which allows knowledge description and implementation, inference with the knowledge, and explanation of reached conclusion in an efficient manner.

2. SYSTEM COMPONENTS

Elementary softwares for testing, preprocessing, reducing, transforming, parametric modeling and for practical information extraction are designed to have high modularity so that data transfer between the elementary softwares can be performed with minimal interference by the analyst. The goal-oriented program is developed interactively on a mainframe computer by following the instructions provided by the knowledgebase system. The concept of the proposed system is illustrated in Fig. 1. Some of the elementary softwares are described briefly in the following sections.

![Diagram](image-url)  
Fig.1 The concept of the proposed system
2.1. A priori tests

Characterization of statistical properties of the objective signal is needed for avoiding the procedural uncertainties cited in 1. Most of the available techniques of random signal processing require the conditions of stationarity and normality. In the case of parametric modeling, additional assumption is further imposed; the signal is assumed to be generated by a linear time invariant system excited by white noise. It is not at all surprising that a technique leads to unreasonable results when these prerequisites are violated. The a priori tests were introduced to measure the extent of violation.

The a priori tests utilized in the present version of the knowledgebase system is moment analysis (Wilks, 1962), zero-crossing analysis (Rice, 1945), Kolmogolov-Smirnov (KS) test (Smirnov, 1948) and sequential probability ratio test (SPRT) (Chien and Adams, 1976). As these tests are well-known and standardized in the area of mathematical statistics, their technical contents are not described in this paper. Previous attempts to utilize the first two tests for characterizing stationarity of time series data can be found elsewhere (see, for example, Kitamura and Upadhyaya, 1982).

The KS test has been employed as a posteriori test to evaluate goodness-of-fit (Upadhyaya and Kerlin, 1978) of a model to the observed signal. The SPRT method has been applied to failure detection in several industrial processes including nuclear reactor (Ray et al., 1983). In this work, however, we try to use them as a priori tests to detect changes in statistical properties of the signal. As we chose to compute the moments up to the fourth order, a total of seven feature parameters is utilized to characterize and monitor the statistical properties of the signal.

2.2. Parametric modeling methods

As the main body of the signal processing programs in the present system development, we focused our scope on parametric modeling algorithms. This is partly because the nonparametric methods such as Fast Fourier Transform (FFT) and correlation analysis are more established than the parametric methods. The other reason is because the number of decision-makings are larger in parametric methods. Selection of model complexity (i.e. model class and order) is a widely-known example of decisions to be conducted cautiously. Choice of modeling algorithm is another critical issue when one is dealing with a signal with nonstationarity and/or pure periodic components. The potential benefit of using the parametric methods mentioned in 1. can be achieved only when these decisions are made in a systematic and rational fashion.

The parametric modeling methods considered in this work are by no means exhaustive, as we treated mainly AR modeling method and a class of ARMA methods. Still we need to consider ample number of methods since there are various optional algorithms for determining the model parameters. Just for the purpose of univariate AR modeling, we have developed five different algorithms as listed in Table-1.

One can use the conventional method of YWAR modeling when the condition of stationarity is satisfied by the observed signal to allow acquisition of sufficient amount of data samples. Otherwise, use of other modeling is strongly recommended since the YWAR method becomes less dependable. As each of the modeling algorithm is characterized by its intrinsic merits as well as demerits, choice of the most suitable algorithm should be made by taking account various aspects of signal properties, purpose of analysis and computational constraints.

The SSLSAR and MEMAR algorithms are, in principle, applicable to data even when the number of samples is too small to be processed by the YWAR method. However, the former requires a significantly large memories. The latter tends to overenhance the spectral resolution. One time-honored procedure of solving this difficulty is to apply one of the methods of trend elimination. If characteristics of the nonstationarity itself (e.g. trend or drift) is regarded to be the quantity of interest, however, use of the TBLSAR modeling is a more appropriate option to meet the need. The memory constraint could be a serious constraint in this case again.

The above paragraph exemplifies only a part of the difficulties encountered during practical analysis of reactor noise signals by means of AR modeling. It can be clearly understood that the decision making becomes far more complicated when one has to treat multivariate signals. Effective utilization of expert's knowledge is a natural and reasonable way to overcome the difficulties.
Table 1. Univariate AR modeling algorithms implemented.

<table>
<thead>
<tr>
<th>Algorithm</th>
<th>Model</th>
<th>Feature</th>
<th>Reference</th>
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</thead>
<tbody>
<tr>
<td>Standard Yule-Walker</td>
<td>YWAR</td>
<td>conventional</td>
<td>(Durbin, 1960)</td>
</tr>
<tr>
<td>Small sample least-squares</td>
<td>SSLSAR</td>
<td>small sample analysis</td>
<td>(Golub, 1965)</td>
</tr>
<tr>
<td>Trend-biased least-squares</td>
<td>TBLSAR</td>
<td>trend model included</td>
<td>(Kitamura et al., 1986)</td>
</tr>
<tr>
<td>Higher-order Yule-Walker</td>
<td>HOYNAR</td>
<td>cancel measurement noise</td>
<td>(Kay, 1980)</td>
</tr>
<tr>
<td>Maximum entropy method</td>
<td>MEMAR</td>
<td>high resolution</td>
<td>(Burg, 1967)</td>
</tr>
</tbody>
</table>

2.3. Noise signature and parameter estimation

The decisions in 2.2. are actually dependent on the quantities to be estimated by transforming the model. Typically, the model is used to derive the characteristics called noise signatures or physical parameters. Some examples are described below.

(1) conventional noise signatures: auto/cross power spectral density (APSD/CPSD) function, ordinary/partial coherence (OCOH/PCOH) function, phase, input-to-output transfer function.

(2) extended noise signatures: noise power contribution (NPC) and signal power contribution (SPC) functions, extended partial coherence (EPCOH) function, inherent noise power spectrum, decomposed feedforward and feedback transfer functions.

(3) physical parameters: signal transit (or delay) time, sensor response time, system stability indices, noise source energy and locations.

While the conventional noise signatures and the physical parameters can be obtained by nonparametric methods as well, the extended noise signatures are usually derived solely by the parametric methods. Although these noise signatures of category (2) are quite powerful in clarifying cause-and-effect relationships between signals, cautious selection of algorithms and validation of the numerical results are imperative. The analysis procedure often becomes recursive, depending on the outcome of the a posteriori tests described in the next section.

2.4. A posteriori tests

It is empirically known that the a priori tests as mentioned in 2.1. are not always sufficient in extracting information for (1) decision making about appropriate choice of signal processing procedures and (2) evaluating physical credibility of the numerical results provided by the methods mentioned in 2.2 and 2.3. In particular, the credibility of parameters or noise signatures estimated through more detailed analysis such as feedback system identification and STP analysis can only be ensured by means of suitable a posteriori tests.

The a posteriori tests in the present program package are; the portmanteau lack-of-fit test (Box and Jenkins, 1971), variance reduction rate (VRR) test (Kitamura, et al., 1979), diagonality test of residual covariance matrix based on orthogonal transformation (Upadhyaya, et al., 1980), and the variation patterns of Akaike information criterion (AIC) (Akaike, 1974) and characteristic roots of AR model (Kishida et al., 1985) for increasing model order. The last two tests were taken into consideration as rich information sources to make decisions about the validity of estimated models.

The fundamental idea of adopting these tests were to characterize the compatibility, or matching, of a specific signal processing procedure with the analyzed signal in a quantitative manner if at all possible. It should be noted, however, that some of the tests can provide us with only qualitative or Fuzzy outcomes related to the required information. The behavior of AIC, for instance, is known to be highly informative for experienced analyst; one can give decisions on appropriateness of the adopted sampling frequency, existence of unawared periodic noise contamination or pure delay component, applicability of the model class, etc. Nevertheless, it is extremely difficult to define a quantitative rule to replace the expert's empirical knowledge to make these judgements. Situation is more or less the same for other tests as well. The development of knowledgebase based on symbolic language is, in our view, one of the most promising means to overcome the difficulty.
3. KNOWLEDGEBASE DESIGN

3.1. Overall structure

It is in principle possible to describe and implement all relevant rules by using the conventional rule representation, e.g. IF (situation is A and purpose of analysis is B) THEN (select procedures X, Y and Z). Such straightforward but oversimplified representation is, however, disadvantageous in comprehending systematic structure of the knowledge, searching out the most appropriate rule, examining mutual consistency of the rules, and in suppressing memory consumption. Sorting and classification of rules are crucial in designing a knowledgebase system.

It should be noticed that each procedure or algorithm is characterized by its intrinsic conditions of applicability as typically described in 2.2. We call this type of rules (i.e. conditions) as procedure-specific knowledge. On the other hand, the objective signals are characterized by their statistical properties and related measurement conditions (e.g. number of bits in digitization, effect of measurement noise, etc.). This class of knowledge is named situation-specific knowledge. The knowledge about the purpose of analysis or parameters to be estimated also belongs to the latter class.

Through examination of the decision-making activities of experts, it is realized that one of the generic tasks (Chandrasekaran, 1986) in noise analysis is to search through the procedure-specific knowledgebase and find out a set of procedures compatible to the signals characterized by the situation-specific knowledge. In other words, the expert must have the heuristic rules about how to relate the situation-specific knowledge with the procedure-specific one. This subset of heuristic knowledge should also be implemented in the knowledgebase independently from the procedure-specific knowledge.

Frame(1):

- parametric modeling
  - slot(1):upper frame value:none
  - slot(2):stability value:good

Frame(1.1):

- AR modeling
  - slot(1):upper frame value: parametric modeling
  - slot(2):correlation calc. value: yes

Frame(1.1.1):

- YW-AR method
  - slot(1):upper frame value:AR modeling
  - slot(2):spectrum resolution value:moderate
  - slot(3):applicability to small sample modeling value:low
  - slot(4):calc. algorithm value:Levinson algorithm
  - slot(5):minimum data number value:512
  - slot(6):memory value:propotional to signal number*2
  - slot(7):valid range of signal status

Frame(1.1.2):

- Least Squares AR method
  - slot(1):upper frame value:AR modeling
  - slot(2):spectrum resolution value:high
  - slot(3):applicability to small sample modeling value:high
  - slot(4):calc. algorithm value:direct fitting
  - slot(5):correlation calc. value:no
  - slot(6):minimum data number value:64
  - slot(7):memory value:propotional to data number
  - slot(8):valid range of signal status

Fig. 2 Example of Frame Representation
3.2. Knowledge representation

The frame representation was adopted to describe the procedure-specific knowledge, while the production rule representation was employed to describe the heuristic rules. The combination of software components consistent with the plant condition and the purpose of analysis is selected based on consultation with the knowledge base system. The consultation is performed by defining the situation-specific information as the input data. In other words, the procedure-specific knowledge implemented by the frame representation is constant and fixed. The dynamic and changeable situation-specific knowledge activates a particular set of production rules to control the strategy of search through the frames.

A part of the frame representation of procedure-specific knowledge is shown in Fig. 2. The tree-type hierarchical structure is convenient to combine mutually related class of algorithms. The implicit function of this representation called "inheritance" is highly useful since the knowledge stored in the higher level frame is automatically transferred to the lower level unless otherwise specified.

For example, most of the parametric modeling algorithms require that the signal should satisfy the condition of stationarity. This procedure-specific knowledge needs to be described only once at the top level by assigning a special value such as "stationarity required" to a slot named "range of signal status". In the SLSAR and TBLSAR modeling, however, this condition is not required. The content of the corresponding slot is thus redefined as "slow baseline change acceptable" in the LSAR frame. The rewritten knowledge is transferred to the frames located lower than the LSAR. The original content of the slot is called default value.

Note that the structure is by no means unique and can be modified depending on the perspective of the system designer. The ARMA modeling algorithm, for instance, may be located at a higher level than the AR algorithm if mathematical generality is regarded as the issue of major concern. In the present design, they are located at two subroots of the same level, because the procedure-specific knowledge on the AR method contains numbers of rules that are irrelevant to the ARMA method; the inheritance function becomes substantially meaningless if the AR frame is located under the ARMA frame.

The production rules for relating the situation-specific knowledge with the procedure-specific one are also organized in a hierarchical manner. A set of related primar rules are grouped as a rule block. This arrangement is also helpful for improving the efficiency of search through the knowledge base. Note that the production rules are extended to cover the outcome of the tests described in 2.1 and 2.4.

Several example of the test-related production rules are given in Table-2 which summarize the knowledge acquired through examining the behavior of AIC. The columns "Observation" and "Suggestion" are related to the condition clause and the conclusion clause of the production rules. Though the rules are defined in qualitative and subjective manner, they can be utilized successfully via interaction with the analyst. There is no necessity to translate the rules into more quantitative expressions. This is of course one of the major benefits of utilizing symbolic programming paradigm. Easiness in adding or deleting the rule is another benefit to be appreciated.

<table>
<thead>
<tr>
<th>Rule ID</th>
<th>Observation</th>
<th>Suggestion</th>
</tr>
</thead>
<tbody>
<tr>
<td>#1</td>
<td>Decreasing smoothly to reach the minimum value.</td>
<td>Adopted model class is valid.</td>
</tr>
<tr>
<td>#2</td>
<td>Decreasing smoothly without reaching the minimum.</td>
<td>Number of samples is large. Reduce it.</td>
</tr>
<tr>
<td>#3-1</td>
<td>Decreasing then resulted in persisting oscillation.</td>
<td>Pure sinusoidal component exists. Truncate the order at moderately high order. Same as above.</td>
</tr>
<tr>
<td>#3-2</td>
<td>Casewise dispersion of opt. order is large.</td>
<td>Model class invalid. Try ARMA model.</td>
</tr>
<tr>
<td>#4</td>
<td>Discontenuous decrease.</td>
<td>Or, possibly same as #2. The signal is white noise. Accept the low order model.</td>
</tr>
<tr>
<td>#5</td>
<td>Decrease is insignificant.</td>
<td></td>
</tr>
</tbody>
</table>
4. RESULTS

Several examples of difficulties experienced in noise analysis and the optional actions suggested by the knowledgebase system are described in this section. Though not exhaustive, these examples would be sufficient to demonstrate the usefulness of the present proposal.

4.1. Effect of baseline shift

First example is to solve the difficulty commonly experienced in estimating APSD of fluctuating components superposed by a significant baseline shifts as as illustrated in Fig. 3. The APSDs estimated by the standard YWAR method is given in Fig. 4(a). The resultant APSD is characterized by the significant increase along with decrease in the frequency. Existence of the baseline shift was easily detected by the moment analysis. The LSTBAR method suggested by the knowledgebase system provides the APSD shown in Fig. 4(b). Note that the effect of baseline shift is eliminated since the trend component and stationary fluctuation are modeled separately in the LSTBAR model. The masking of some of the peaks in the APSD of Fig. 4(a) was successfully avoided by the optional procedure.

Fig.3 Time record of test signal with linear trend.

Fig.4 PSDs of sample data:
(a) by YW-AR method,
(b) by Least Squares TBAR method.
4.2. Effect of sinusoidal noise contamination

The effect of sinusoidal noise contamination is sometimes disturbing in automatizing the reactor noise analysis. As mentioned in rule #3-2 of Table-2, the simple automatic determination of the optimal AR order can often lead to large casewise dispersion, causing significant casewise variation in APSD pattern which is not real from physical viewpoint. The casewise variations of optimal order determined by the AIC are summarised to the histograms shown in Fig. 5(a) and (b) for the same signal with and without noise contamination, respectively.

The APSD obtained by following the suggestion of rule #3-2 is given in Fig. 6. In addition to the global structure, the peak introduced by the contamination component is clearly reproduced in this result. The peak becomes unnoticeable because of degradation in spectral resolution if a model order as low as six or seven is adopted. Though the global APSD structure is still maintained in the lower order model, the disappearance of the sharp peak can cause serious difficulty in an automated diagnosis system based on APSD pattern recognition of reactor noise. This potential danger is successfully avoided by introducing the suggestion.

Fig. 5  Histogram of optimal order selected by AIC for the signal
(a) with noise contamination
(b) without noise contamination

Fig. 6  APSD obtained by following the suggestion of rule #3-2

Fig. 7  Behavior of the characteristic roots obtained for ARMA noise.
(Circles represent robust roots)
4.3. Validity of AR model

The difficulty in model order determination as mentioned above is, in general context, attributed to the fact that the signal generating process is hardly approximated by the AR model. The ARMA process is another typical example of such processes. Under certain pole-zero assignment condition, it is quite difficult to approximate the ARMA process by the AR model. The examination of the AIC behavior is not sufficient to overcome the difficulty.

The examination of characteristic roots of the AR model is highly helpful in evaluating the validity of the AR model. Locations of the characteristic roots in a complex plane defined in terms of $Z^{-1}$, i.e. the backward shift operator are illustrated in Fig. 7 for increasing model order. The semi-circles with increasing radius correspond to the unit circles for increasing model order as denoted.

As reported earlier (Kishida et al., 1985), two types of poles called robust and nonrobust pole are observed. The robust poles represented by the open circles are observed at the same locations, while the nonrobust poles change their locations significantly. Existence of the nonrobust poles indicates that the process is hardly approximated by the AR model. It is not unusual that the AR model order becomes unacceptably high to reproduce the ARMA spectra. In other words, the AR spectrum is only a poor approximation of the ARMA spectrum as far as practically-acceptable order is adopted. The examination of characteristic root as shown in Fig. 7 is highly helpful to overcome such problems.

5. CONCLUDING REMARKS

Design philosophy and current status of our knowledgebase system for generating goal-oriented noise analysis softwares are briefly described with several illustrative examples demonstrating the potential of the proposed method. Though the numerical results shown in 4, are restricted to univariate data, the knowledgebase is covering the expertise for overriding difficulties encountered during multivariate analysis as well. Furthermore, the system is designed to allow easy updating of the contents; the knowledgebase system can be revised base on field experiences. The present system can be utilized as an assistance tool for developing noise analysis softwares and for educating unexperienced analysts.

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AN EXPERT SYSTEM APPROACH TO THE DEVELOPMENT OF NOISE DIAGNOSTIC SYSTEM IN NPP PAKS

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Abstract - The paper describes the approach applied in the development of a knowledge based reactor noise surveillance and diagnostic system. The emphasis is taken on the discussion of the signal processing subsystem where declarative and procedural knowledge is combined. The numeric procedures included are parametric AR, ARMA time-domain and related frequency domain methods. The functions and global architecture of the system is described.

Keywords - reactor noise; AR, ARMA models; signal processing; knowledge based systems.

1. INTRODUCTION

Noise surveillance and diagnostics in Nuclear Power Plants /NPP/ has now a more than twenty years old tradition. Processing the randomly fluctuating component of NPP signals, several methods have been elaborated to diagnose abnormal behaviour of plant systems or to detect failure of components.

Applying the methods of noise diagnostics, many important relationships could be established e.g. among the spectral characteristic of noise signals and the operational status of the plant and its components. It was proved that changes in viability and in physical parameters could be detected in an early stage of their appearance, which was a significant contribution to reliable operation of the power plant, see Bernard at.al (1986) for a "state of art" on this issue.

For the above reason serious effort has been directed to develop reliable automated diagnostic systems that can be used in operating plants, and where the emerging technologies of pattern recognition and expert systems are fully utilised.

This paper investigates the concept and elaboration of the noise diagnostic expert system developed for NPP Paks. The subsystem designed for time- and frequency domain analysis of noise signals /signal processing subsystem/ will be discussed in more detail.

Conventional methods for processing the noise signals are based on the estimation of spectral densities using the Blackman-Tukey or variations of Fast-Fourier /FFT/ methods. It can be deduced that these /nonparametric/ spectral estimates ensure limited possibilities for detecting changes in the noise signal characteristics.

In our approach to the noise diagnostic system, a so-called "method - base" consisting of /besides the nonparametric methods/ various parametric spectral estimation methods associated with AR, ARMA and transfer matrix modell structures was elaborated. The method-base includes all the off-line recursive structure and parameter estimation and also adaptive procedures needed for the identification of the above models.

Using the reference model-library or model-base, it becomes possible to apply pattern recognition to detect significant changes in the model structures and parameters in the parameter space. Based on the identified parametric models, the spectral densities are also calculated and a subsequent recognition follows in the frequency
domain. The STP and feedback analysis is also performed here.

The detected significant changes are evaluated and forwarded to the knowledge base
of the diagnostic subsystem that can associate them with plant anomalies.

The operation of the signal processing system is controlled by a signal processing
supervisor which is a shallow coupled knowledge-based system combining symbolic
and numeric computation.

The functional structure of the system is reviewed in the 2nd paragraph. The func-
tions of the signal processing supervisor and the approach applied to design its
architecture is discussed in paragraph 3.

2. THE FUNCTIONAL STRUCTURE OF NOISE DIAGNOSTIC SYSTEM

The main functions of noise analysis techniques applied in NPPs can roughly be
devided into three categories: surveillance, pattern recognition and diagnostics.
The surveillance of the system is based on the measurement of process signals and
their conversion into "usable" forms in which the relevant signal properties
can be distinguished. This task requires the application of a broad range of sig-
nal-processing methods. This is followed by a pattern recognition where the usual
classification "normal" and "abnormal" associated with the status of the plant or
of the components of the plant is performed. The diagnostics problem is concerned
with identifying the source and degree of the detected anomaly.

The functional structure of the NDS satisfying the above three tasks is illustrated
on Fig. 1.

The subsystem represented by Block 1 is a knowledge-based signal processing super-
visor /SPS/, its functions and realization will be discussed later.
The Block 2 subsystem performs data acquisition and signal preprocessing. Control-
led by the SPS, it can perform the following tasks:
- Goal or problem driven selection of signals to be measured, with specified sam-
ping frequency and gathering given number of samples.
- Preliminary statistical analysis of data: tests for outlier detection, station-
arity, need for differencing, e.t.c.
- Applies prespecified transformations /normalizing, trend elimination/ and
prefiltering of the data.
Other user defined possibilities provided by this subsystem is the nonparametric estimation of statistical functions, like auto-, cross- and partial auto-correlation functions and FFT based spectral densities. The selected and preprocessed signals are allocated into a specially organised file system called signal library, which is input to the signal processing system. These functions are illustrated in Fig 2.

The main function of the signal processing system is the identification of parametric like AR, ARX, ARMA, ARMAX or transfer function models /see Bokor and Kevicsky (1982) and Gevers and Wertz (1984) for details/ using the samples in the signal library. The organization is illustrated in Fig 2. The system operates on the method base, which is a collection of procedures for structure and parameter estimation of the above models. A short description of the methods available in the method base is described in Nagy et al (1987). The selection of a specific method is controlled by the SPS. For the sake of avoiding "prejudices" induced by the necessary preassumption on applicability of a given method, more than one methods are applied to solve the same task, and the results are compared. The model validation is based on specific tests, applied to the residuals. An accepted model of the signals is allocated into the model library, which is the input to two subsequent subsystem: to the subsystem for feature extraction and change detection in the parameter space and to the parametric spectral analysis subsystems.
The parametric spectral analysis subsystem performs the computation of power spectral density /PSD/ and associated functions from the identified models. It is also possible to apply Signal Transmission Path /STP/ and Signal Effect /SE/ analysis, as described in Veres et al. (1987).

The specific features characterizing the most important properties of PSDs are also computed and the results are allocated in the spectral library. The concepts for defining features /discriminants/ for PSD will not be discussed here see e.g. Piety (1977) for some typical definitions. Now, the meaning of transforming the process signals into "usable" form can be explained more precisely: these forms are the models of signals in the model library and the spectral characteristics /features/ allocated in the spectral library.

The next step is the detection of changes, when comparing these models, spectral functions to those accepted to represent the "normal" status. This problem can be solved both in the parameter space defined by the identified models /Block 4/, and in the frequency domain, using appropriate features of the spectral functions /Block 7/.

The change in the "normal" status can be detected in the parameter space by testing (i) the change in the model structure /McMillan degree of AR, ARMA models, or maximum delays in AR or MA operators/, (ii) change in parameter estimates. The solution of problem (i) is supported by applying specific tests on the model structure, see e.g. the Lagrange-multiplier and Generalised Likelihood Ratio /GLR/ tests. The problem (ii) can be solved by applying discriminant analysis in the parameter space.

Discriminant analysis can also be applied to detect changes in the feature-space, but in many cases the use of simple statistical tests is also convenient, see e.g. Smith and Gonzales (1985).

Block 8 represents the diagnostic subsystem, where the detected changes are interpreted and related to the source or cause of suspected anomalies.

This is a typical knowledge-based systems, where the results of pattern recognition is combined with the declarative knowledge /rules and facts/ available in the knowledge base.

The function of the subsystem denoted by Block 5 is the detection of abrupt changes in the system. This function requires on-line data processing, and the construction of special filters /e.g. the Kalman-filter/ representing the model of "normal" status. The changes can be detected by defining appropriate statistics /based e.g. on the innovation sequence of the Kalman-filter/ and decision rules, see e.g. Baseville and Benveniste (1986) for a summary of change detection methods like the multiple model or generalised likelihood ratio approach.

In our approach the filters are constructed from the reference models allocated in the model-library. When significant change is detected, an alarm is forwarded to the SPS, which can initiate a detailed analysis of the new status and the diagnostics of the identified changes.

### 3. THE KNOWLEDGE BASED SIGNAL PROCESSING SUPERVISOR

The main function of this subsystem is the control of the communication with the operating personnel and the control of operation of the data acquisition and signal processing subsystems. It operates in two basic mode: in the learning and in the diagnosing mode.

The functions in the learning mode are the following.

(i) Assist the operator personal to manage the signal processing task and to interpret the results.

(ii) Build up the library of reference models and spectral function representing the "normal" status.

(iii) Assist the choice of specific features used in the pattern recognition and diagnosis.

In the diagnosing mode the operator specifies the goal /e.g. monitor the status of a given component/, and the SPS assists to perform the following tasks:

(i) Prescribes the selection of signals to be measured in the data acquisition system.

(ii) Describes the experimental conditions to be applied.

(iii) Selects the methods for signal processing, spectral analysis and feature extraction.

(iv) Forwards the results to the pattern recognition and diagnostic system.
The SPS is a knowledge based system, where symbolic and numeric computation are involved. Declarative knowledge represented in the knowledge base are e.g. the knowledge about the sensitivity of various signals on the dynamic modes of main components, and the relation of physical phenomena to the selection of sensors. In addition, the inclusion of knowledge about the numerical methods embedded in the signal processing system and reasoning about the utilization of their results is also relevant. Thus the SPS was designed to be a shallow coupled symbolic and procedural knowledge based system, see Kowalik (1986) for various properties of such systems. The problem solving procedure is controlled by defining problem state-variables and rules describing relationships between these state-variables and their acceptable values. The communication between symbolic and numeric processes is ensured by simple data record exchange. The typical function of the symbolic processes are the user interface, interpretation of the computed results and control of the problem solving process. The functions of the numeric processes involved in the signal processing subsystem were characterised in the previous paragraph.

The system was realised on an IBM PC/AT compatible computer. In the present form the SPS is written in PASCAL, and the numerical processes are written in FORTRAN. Recent development is going on using PROLOG and applying commercial expert system shells.

4. CONCLUSION

The paper described the approach applied in the development of a knowledge based reactor noise surveillance and diagnostic system. The emphasis was taken on the discussion of signal processing subsystem, which is a shallow coupled system combining symbolic and numeric computation. The focus in the elaboration of the method-base was on the application of parametric time- and frequency domain methods based on AR, ARMA modelling and identification of noise processes.

The described system was designed as part of the noise surveillance and diagnostic system operating on the 3rd and 4th reactor unit of NPP, Paks. It is applied for noncontinuous surveillance and operates mainly in learning mode to build up the reference model- and spectral-libraries, and also the associated knowledge bases. The results indicate, that considerable effort is still needed to identify the appropriate surveillance and diagnostic parameters for each signals before the system can be made more automated.

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OPERATING EXPERIENCES WITH AN ON-LINE, COMPUTER BASED NUCLEAR PLANT SURVEILLANCE AND ANOMALY DETECTION SYSTEM BASED ON PATTERN RECOGNITION AND ARTIFICIAL INTELLIGENCE

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ABSTRACT

The control room of a nuclear power plant can represent a hostile work environment for all but a highly trained operating team. Whilst disciplined training and long professional experience will guarantee assurance of safety and reliability in nuclear plant operation, any surveillance system which has the ability to minimise human error and provide additional safeguards is a desirable asset. This paper proposes a scheme whereby some key parameters of a nuclear power plant, precisely known through detailed calculation and accurate measurement are stored as a data base in an on-line computer. Through the systematic statistical analysis of key stochastic variates a comparison is made by the on-line system with the data base at regular intervals. These time intervals may be as short as seconds during periods of reactor transients such as at start-up or shut down. Alternatively, during steady state operation, the parameters are calculated and displayed at intervals of an hour or greater. An anomaly, or an indication of unusual operational behaviour is indicated both numerically and graphically by the computer if it detects a variance greater than a few percent from the mean value of the reference data base.

KEY WORDS

Stochastic variate, statistical analysis, autoregression analysis, anomaly detection, surveillance, pattern recognition, expert system, artificial intelligence, on-line computer, reactivity, eigen-function, eigen-value, mechanical vibration, feed-back coefficients, decision making.

INTRODUCTION

Artificial intelligence (Reference 1) is concerned with enabling computers to imitate the characteristics that make people seem intelligent. This statement raises the question of just what is meant by human intelligence.

Because the answer to the above question is so difficult, especially within the context of those who have to interpret and respond to the instantaneous states and dangers in the reading of instruments in a nuclear power station control room - see Appendix - it is perhaps better to avoid the answer. In fact the use of the concept, because it is so ill defined, might lead to fallacious answers. For this reason, we prefer to define the present stage of development of our computerised nuclear power plant surveillance systems to that of an "expert system". In this way we can avoid the vague statements of the type that "intelligence appears to be an amalgam of many different information processing and information representing capabilities. Within such a statement is implicit the concept that information is communicated knowledge. At this stage, perhaps the only aspect of artificial intelligence inherent in the "SNEDAC" System (School of Nuclear Engineering Data Acquisition Computer) is the response function of scrambling a reactor when six out of ten key parameters which are sequentially displayed on a Control Room V.D.U. screen display diagnostic responses statistically out of keeping with what is deemed to

* The author is a Foundation Member of the Australian Nuclear Association, a Fellow of the Australian Institute of Energy and Chairman of the National Panel on Nuclear Engineering of the Institution of Engineers, Australia.
be the prudent operation of a multi-billion dollar nuclear plant. As SNEDAC in 1987 dollars is valued at approximately $350,000 it is deemed to be a prudent investment.

Details of SNEDAC from a purely technical point of view have been presented at Gainesville, Florida, in 1963 and 1965 and at Casaccia (1974), Gatlinburg (1977), Tokyo (1981) and Dijon (1984). In a hierarchical sense it is possibly the first of the large data base - twenty channel - nuclear plant monitoring systems even constructed and its capabilities since the late 1950s have been continually upgraded to the extent that the author is now prepared to call it an "expert system".

The unique features of SNEDAC from its earliest days can be summarised as follows:

(1) The expert system is independent of nearly all of the standard Control Room instrumentation channels with their confusing multiplicity of sizes, geometrical shapes, cluster assemblies, calibration facilities and the almost deliberate attempt implicit in their design to make the man-machine interface as difficult to interpret in the cognitive sense as possible.

(2) SNEDAC relies on independent transducers, power supplies, signal channels and numerical-graphical display which qualify it to be classed as an "expert system".

(3) SNEDAC's display to an operator whose capabilities of interpretation of graphics and numbers are limited are enhanced by providing a facility which, in answer to the question "is your system operating in an anomalous manner" is a simple clearly understandable "yes" or "no".

Behind such a simple concept of an expert system is twenty years of hard work ranging from major off-line computer calculations and sophisticated three dimensional experimental measurements, the results of which form the data base of the expert system operation. This will be discussed briefly in the subsequent section.

PLANT SURVEILLANCE

As a theoretical concept the HIFAR reactor has been geometrically subdivided into twenty three autonomous regions which make allowance for nineteen fuel elements. Neutron transport and neutron diffusion theory has provided a reliable theoretical base for eigenfunction "shapes" and eigenvalue "parameters" which form one segment of the database in the dedicated computer. A second vector of normally observed key parameters is obtained from precise measurements arising from:

(1) Eight neutrons
(2) Four temperature
(3) Two fuel structure
(4) Two vibration
(5) Two mass flow

channels continually updating the computer with carefully controlled digital and analogue signals.

From this experimental flow of data the expert system - independently of a clutter of control room instrumentation, can display at call, up to ten - see Figures - vital and easily interpretable set of numerical/graphical displays which independently assume the operator of system safety. If this staff member is diffident about responding to such a display he can always revert to the "yes" or "no" sequence. The displays range from percentage changes in the flux and gamma distribution across the reactor core in the global sense to direct indications of sub-critical through critical indications of reactivity, temperature coefficients of reactivity and void coefficients to measurements of burnup in the reactor core - most of which can be "dialled-up" at remote distances from the power plant - if necessary in a centralised Nuclear Regulatory Commission Control Office or a smaller Utilities' Command Centre. In essence therefore, we have succeeded in replicating by perhaps two orders of magnitude the understanding of the normal or anomalous behaviour of nuclear plant operation. This is not to say that the thousands of instruments, chart recorders and multi-faceted dials and control knobs in the surgically clean control room have become redundant.

The only claim made for this example of an expert system is that the possibility of a second Chernobyl where such instrumentation is available becomes vanishingly small. However, the proviso for reliable containment has not been abrogated!
What is the generic, hierarchal structure of such an "expert system"?

THE EXPERT SYSTEM APPROACH

It is characteristic of immature fields of scientific research that the specialists are unable to agree on a definition of their chosen field. It is axiomatic that "expert systems" fall into this category. The Appendix goes into a little further detail of Expert System definition but it is appropriate that at this point we list our own essential attributes to the field we have inaugurated for nuclear plant.

As a multiple input-output device the reactor has already been identified in terms of cross correlation matrices and power spectral densities with a high or weak degree of correlation between some of the variables already cited.

For the structure of an expert system identification and ultimately an artificial intelligence mode of control of such plant which has a great potential for over-ruling human error - whether due to negligence, complacency or down-right poor design of control systems we need to utilise

1. Backward/Forward chaining
2. Bayesian probability
3. Variance analysis
4. Empty shell
5. Fuzzy logic
6. Heuristics
7. An inference engine - hardware and software
8. Knowledge base
9. Knowledge engineering and
10. Graphical or pattern matching

In a language more fit for sane human beings - of whom an extraordinary percentage are nuclear engineers and scientists, we need simply to correlate the following procedures:

1. From reactor physics hyperfine structure and global computations we need to extract a set of numbers germane to the power system design and utilising neutron transport and neutron diffusion codes. (My favourite and fervent hope is that our profession will eventually press very hard for global standardisation in this and the following areas.)

2. The second stage is to incorporate from a combination of theory and experiment, another set of numbers which make allowance for power phenomena and negative or positive feedback at the exact locations where our "noise transducers are located.

3. Having reached this point, we can decide on the most appropriate data base for our on-line computer.

4. This data base is now ready to become an evolutionary source of reference and comparison for the operating plant and finally

5. A decision making statistical control programme can now be written to present standardised numbers and graphics for the production of on-line operational coding which is easily recognised by say, a shift manager as an over-riding control factor eliminating on an independent basis the confusion producing diversity of control room instrumentation. The algorithms, compilers, decision trees and control flags for this system are being published separately elsewhere due to length considerations.

CONCLUSIONS

At this stage whether the expert system wins its battle against the highly intuitive emotional and introspective logic of the expert professional will still be in the hands of a brilliantly endowed but fallible human being. As of September 1987, the game play against the latter is 0-3 (at least). Hence one can conclude with great optimism that this confidence is not in vain and that within one or two decades the professionals may to some extent withdraw from the competition as the Windscales Three Mile Islands and Chernobyls become less probable.
ACKNOWLEDGEMENTS

The author is deeply grateful for useful discussion with his colleagues at the University of New South Wales, the Australian Nuclear Science and Technology Organisation and the International Atomic Energy Agency.

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APPENDIX (Reference 2)

INTRODUCTION

At its present stage of development, "Artificial Intelligence" is the pet and current jargon of the computer scientist and meaningless to the practising engineer or technologist. It is an ill defined concept, or to be more topical, a "fuzzy system".

To a large extent, Artificial Intelligence is speculatively based on a cognitive process of interaction followed by a response or a control signal from a complex, multi-tier level of computer programmes operating on an extensive data base. From this point of view, the conversation between operator and computer - the man/machine based interface - can in principle, be given an "artificial intelligence" which can over-rule the human thought process under all circumstances. Such a facility may not be desirable when the data base originates from a system as complex as a nuclear power reactor or in fact any component part of a nuclear fuel cycle or a nuclear fusion cycle.

EXPERT SYSTEMS

From a pedantic point of view, therefore, the system developed by the author is an "expert system" in which the parameters and profiles related to the safety of a nuclear power station and "keyed-in" to both core and mass memory of a computing system and represent an idealised data base to which reference can be made, as frequently as desired, by an up-dated set of parameters systematically retrieved from the operation of the unit. The "expertise" of the computer is derived in this instance from a process of statistical algorithms which can display after computation both numerical sequences and graphics. Typical examples of how a set of such numerical-graphical combinations can give a simple cognitive description which can attract both the visual and the auditory senses of an operating team in a control room have been given in the main text.

ANOMALY DETECTION

The "fine tuning" of the computer based expert system can be based on, say, a five percent deviation of one out of six key parameters. Its consequence is planned to be a colour change from red to green on a VDU screen, together with an audio signal. Both the numerical and graphical sequences displayed are cognitive with respect to easy recognition of normal operating conditions and instantaneous response of an anomalous state in

Flux distribution (neutron and gamma)
Reactivity
Temperature
Void effects and flow conditions
Core barrel vibration or acceleration
Fuel element fine structure parameters

COMPONENTS OF THE EXPERT SYSTEM

The design of the system described in this paper requires the following components:
DIFFERENCES BETWEEN CONVENTIONAL AND EXPERT SYSTEMS

Conventional systems

. Contains orderly and deterministic processes.
. A single input goes through a single mechanism or algorithm to produce a correct output.
. Program code and data are kept separately. The recipes for manipulation and the structure of information are intermixed in the code.
. The embedded manipulation and structure makes it difficult to modify complex systems.
. No reasoning or explanation given to the user about why a particular input resulted in a particular output (except through a roundabout mechanism of program trace in debugging tools).

Expert systems

. Contains heuristic and rule of thumb processes.
. Multiple common (often redundant) inputs go through overlapping mechanisms to produce multiple plausible solutions.
. The manipulation rules are held separately (in the inference engine) from the structure of information (in the knowledge base).
. A separate knowledge base can be amended relatively easily.
. The knowledge base is intentionally made visible to the user.

Fig 1: Generation, Propagation and Dispersion of Nucleate Boiling at 10 M.W.
Fig. 2 Normal transfer and global coherence function of just critical reactor.
Fig. 3 Pattern display of 1mm dia. helium bubble moving past miniature fission chambers.
Fig. 4 Anomalous behaviour of control system hydroelastic vibration.
Fig. 5 Raw neutron noise data and computerised sampling system
16, HANNING... 0
17, FRAME SIZE... 1024
18, SAMP. FREQ... 2
19, FREQ. FACTOR... 2

110, ATTN. CODE... 55
111, BUFFER CODE... 0
112, TRIG. CODE... 0
113, FILTER CODE... 0
114, NO. FRAMES... 10
115, CH. CODE... 3

Fig. 6 Remote sensing of system reactivity and fissile inventory.
Fig. 7 Printout from thermocouple instrumental fuel elements.
### Polynomial Regression for Dependent Variable TSI

#### Matrix Test I

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### Coefficients of Orthogonal Polynomials in Variable TSI

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**Fig. 8** Non-linear autoregression analysis of normal and anomalous hyperfine structure.
Fig. 9 Off-line computation of voidage effect on reactor fluxes.
Fig. 10 Out of core vibration measurements.
MECHANICAL AND PROCESS MODELLING

Session chairman: G. Hughes (U.K.)
SUMMARY OF THE SESSION

Bauernfeind reported a model of a four-loop PWR primary circuit. The model works in the frequency domain and allows the computation of vibrational power spectral densities. It was used to calculate the influence of mechanical degradation and damage on the functions monitored by a vibration monitoring system. Good agreement between model and measurement was demonstrated.

Shinshin presented calculations of neutron flux perturbation due to control rod vibration using a two energy-group diffusion model. The model was applied to a PWR control rod element to demonstrate that malfunction detection and location can be achieved using ex-core detectors. The work was confirmed by experimental observations.

Upadhyaya described a detailed theoretical analysis of the cross power spectral density phase relationships between in-core flux changes and core exit temperature fluctuations. Modelling applied to the LOFT reactor showed a linear change in phase over a frequency range of 0.1 - 2 Hz. The influence of the moderator temperature coefficient of reactivity is discussed. The analysis further showed that coolant flow rate fluctuations are the primary driving force, which was confirmed by independent studies. It is suggested that phase behaviour is a good way of monitoring the moderator temperature coefficient.

Antonopoulos-Domias explained the structure of neutron noise coherences at low frequencies in BWR’s, drawing attention to the noise in coherences at very low frequency. A 1-D, two region (fuel-coolant) reactor model is used to demonstrate that pressure feedback could be one reason for the rise in coherences, which are observed experimentally. Other experimental evidence of another mechanism is satisfactorily modelled by the use of space-dependent feedback.

Jena reports Indian work on transport theory predictions of the transmission of neutron noise through non-multiplying media, with the objective of estimating the performance of ex-core detectors on fast reactors. Both infinite and finite media cases are considered; the latter providing a more realistic estimate of the upper break frequency, which is calculated to be of the order of 10 Hz for transmission through graphite compared with 100 Hz for borated concrete.

Messingual-Bruynooghe demonstrated how in-core neutron detectors can be used to identify and locate vibration phenomena, including fuel, control rod and thimble movements in 900 and 1300 MWe PWR's. In addition fluctuations caused by local boiling are considered. Despite the restricted spatial sensitivity of the detectors it is possible to detect remote perturbations in the frequency range within the plateau region of the reactor transfer function, and obtain reasonable spatial location. It is then possible to decide the nature of the anomaly by comparing theoretical predictions with measured values.

Nomura presented a method of monitoring reactor systems by a state space trajectory pattern. The method has been applied to one and two dimensional equations and to reactor systems with temperature and void feedback using a computer simulation. The trajectory patterns permit direct treatment of new data and can cope with non-linear systems. It is proposed to utilise pattern recognition methods in n-dimensional space to automatically detect anomalies.

Kosma outlined his study of a space-dependent coupled neutronic-thermohydraulic noise model. The eigenvalues were shown to be a useful method of understanding feedback effects. Critical combinations of coolant velocity and heat transfer coefficient were explored for a PWR core resulting in low frequency resonances (0.15 - 0.4 Hz) which are an inherent feature of light water reactors.
VIBRATION MONITORING OF A FOUR-LOOP PWR: MODEL-INVESTIGATIONS OF THE SENSITIVITY OF THE MONITORED SIGNALS ON MECHANICAL FAILURES

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ABSTRACT
Since years Vibration Monitoring Systems (VMS) are used in German PWR with more and more success. Experience has been gained with respect to detect mechanical failures at a very early stage of development.

However, the influence of incipient failures on the monitored vibrations of the primary components are known only for few cases, when mechanical failures occurred at PWRs in reality and when at the same time a VMS was in operation there.

In order to get more information on the sensitivity of the monitored vibrations on failure caused changes in the functions or patterns deduced from the VMS signals, model investigations including mechanical failures have been performed. As a first step an analytical vibration model has been formed, which can calculate "analytical vibration patterns", matching the "measured vibration patterns" quite well. The monitored functions are the APSDs of the VMS-signals.

Using this model it was possible, to change certain stiffness and damping parameters in such a way, as we assumed that any mechanical failure could happen in a real structure. By calculating the analytical vibration patterns using the failure model and by comparing these results with the reference patterns, the changes of the monitored signals caused by the failure in question could easily be evaluated. In this way also the effects of a growing failure like more and more relaxing hold down springs at the pressure vessel flange were investigated. In addition, constrained supports of components or internals and broken tie-rods of steam generator or main coolant pump supports have been simulated.

A great number of incipient mechanical failures and breaks of supports were such investigated with respect of their effect on the different VMS-signals and patterns respectively. The calculations were performed for the primary system of a 1300 MW el. power PWR with four loops. The work was sponsored by the Minister of Research and Development of the FRG.

In the paper an overview of the results as well as some examples with details of interest will be given. In addition, the most important lessons learned from these sensitivity investigations will be presented.

INTRODUCTION
For a couple of years Vibration Monitoring (VM) has been used with success in German PWR's. Noise specialists as well as utilities have gained more and more experience with this relatively new technique. Advanced VM-systems, based on data bank and pattern recognition techniques, are being developed to get still more and better information on the mechanic state of the monitored structures.
However, one kind of information can hardly be obtained by trial and error, viz. the information about the sensitivity of the system to the different kinds of mechanical failures. The first reason for this is that - in terms of statistics - only a small number of VMSSs are in operation in West Germany. The second reason is the small number of mechanical failures, which have occurred up to now at the primary systems of the PWR's and - of course - we hope that this will be so also in the future.

On the other hand, for practical application of VMSSs the plant operators need alarm thresholds. In a similar way, also the future computer based VMSSs will need correct threshold values to give alarms in time but to avoid false alarms. One precondition for correct thresholds is to know in advance how certain mechanical failures will change the monitored patterns.

THE ANALYTICAL MODEL

Of course, the information needed cannot be obtained by experiments at operating plants. The best way - now as before - is to use analytical models, describing the in-operation vibration behaviour of the monitored structure.

Such a model has to offer the possibility of calculating in an analytical way the vibration patterns, which correspond to the measured VM-patterns. The patterns predominantly used are the Auto Power Spectral Density (APSD) functions of displacement, neutron noise and pressure fluctuation signals.

Therefore, the model (fig. 1) developed at the GRS in cooperation with KWU, works in the frequency domain /1/. It calculates the APSD's of the vibrations of the different degrees of freedom caused by a complex excitation matrix. Subsequently, a transformation is performed in the way shown on top of figure 2 so as to obtain what we call the "theoretical measurement spectra". An example is shown in figure 3. The model calculates the vibration of the pressure vessel, i.e. the PSD of its rotation S_{KV} and the PSD of the vertical vibrations S_{KV}. Via the transformation procedure, the displacement spectrum resulting from the rotation and the vertical movement is calculated for exactly that point of the RPV, where the absolute displacement sensors are located.

---

**Calculation of theoretical vibration signals**

\[ S_{KK}(w) = \sum_{i=1}^{N} \sum_{j=1}^{N} \hat{g}_{ik}(w) \hat{g}_{jk}(w) S_{ij}(w) \]

- \( S_{KK} \) = theoretical measurement spectrum
- \( S_{ij} \) = calculated output spectrum
- \( \hat{g}_{ik}, \hat{g}_{jk} \) = frequ. response function for the transformation

---

**Fig. 1: Finite Element model of an PWR 4-loop primary system**

**Fig. 2: Steps of model-calculations to obtain theoretical vibration spectra**
The model is based on Finite Element Technique. To minimize CPU-time and, accordingly, computer costs, the number of degrees of freedom is chosen as small as possible, which means that only those vibration modes are modeled, which are excited during plant operation. The number of degrees of freedom needed is 28. One loop and the RPV and its internals are modeled in detail. Three loops are taken into account by modified supporting conditions of the pressure vessel.

The excitation functions are defined using on-site pressure fluctuation measurements, investigations of scaling models and - as far as correlation and phase relation of the forces are concerned - analytical considerations /2-4/. The spectral composition of the excitation is based on the jet-noise set up combined with forces caused by standing waves inside the hot and the cold leg of the primary system.

Two "theoretical displacement spectra" calculated by means of the model are shown on top of figure 4. The lower part of the figure shows the corresponding measured spectra A1 and A2. As has been mentioned before, 4 absolute displacement sensors are mounted on the flange of the pressure vessel head, each of them shifted by 90°.

The figure shows quite clearly that, for the most important vibration modes, a correspondence good enough for sensitivity studies is achieved.

\[
S_{A2,A2} = a^2 S_{PP} S_{VV} + 2a \text{Re} \left\{ S_{PV} \right\}
\]
\[
S_{A4,A4} = a^2 S_{PP} S_{VV} - 2a \text{Re} \left\{ S_{PV} \right\}
\]

**Competition of theoretical measurement signals**

*Fig. 3: Calculation of theoretical absolut displacement signals of the pressure vessel*

*Fig. 4: Comparison of two theoretical and two measured absolut displacement spectra*
SENSITIVITY STUDIES
The aim of the studies was to find out the influence of incipient mechanical failures on the functions used in a VMS. Slowly developing failures, such as relaxing hold-down springs or stiffening junctions or supports of the components as well as the influence of damage, such as broken tie rods at a Steam Generator (DE) or at a Main Coolant Pump (MCP) have been investigated.

In the model the stiffness and damping parameters of the component parts affected by the failure have been changed. That was done step by step to find out the most sensitive degree of freedom and, accordingly, the most sensitive VMS-signal for each failure under investigation.

Two examples of investigated failures will be discussed now,
- failures of steam generator (DE) supports,
- degradation of reactor internals (RI) hold-down springs.

For identification purposes, each peak in the spectra has got a number (table 1). In this table also a short description of the vibration mode and the measured resonant frequency is given. Three spectra are shown for each investigated case, at the upper right of the drawing the reference spectrum, at the upper left the spectrum with the mechanical failure in question and at the bottom both spectra within the same figure.

In the case of failures at DE-supports, we assumed that at the beginning the DE will be fixed in one radial position at its top. At the measurement position RIV at the hot leg, primarily the rotation and the vertical translation modes of the DE itself (0, 8, 17, 18), but also the vibration of the MCP were influenced by the failure (figure 5). The same changes can be observed at the pressure vessel vibration signals (figure 6). Of course, the changes are not as significant as in the spectra of the affected loop, since the pressure vessel is influenced by three sound and one defect loop.

Fig. 5: APSDs of the relative displacement signal - Baseline and Steam Generator top constraint

Fig. 6: APSDs of the absolute displacement signal - Baseline and Steam Generator top constraint
If, as a consequence of the proceeding damage one of the tie-rods supporting the DE brakes, we get severe changes also in the RPV vibration spectra (figure 7). For example, the steam generator rotation frequency of 26.8 Hz goes down to about 7 Hz and coincides with the resonance peak No. 8 of the MCPs. The consequence, of course, is a strong increase in the combined peak due to the resonance effect. However, the peak 17a decreases in comparison with the case fixed DE top. Peak 13, finally, the vertical vibration of the steam generator drops to about 10 Hz and comes into the frequency region between the pendular modes of the core (10) and the core barrel (12).

The second example comprises clamping conditions of RPV-internals or degradations of internals themselves. We investigated cases like contact of pressure vessel - core barrel at bottom grid plate, spring degradation of the grid plates, degradation of hold-down springs of the fuel elements, inclination of the upper core support structure and degradation of the pressure vessel hold-down springs.
These hold-down springs are installed in the flange of the pressure vessel top and press down the flanges of the upper core support structure (OKG) and of the core barrel (KB). The influence of increasing spring degradation has been investigated for both components alone and in combination.

Figure 8 shows the case of spring constant reduction of the core barrel spring in steps of about 10%. The most affected modes are the core translation (10) and the RPV rotation (22). Decrease of the rotational spring constant of the upper core support structure causes above all changes of peak 12 characterizing the core rotation mode, whereas the core translation frequency 10 doesn't change at all (figure 9). If we diminish both spring constants (KB and OKG) then, of course, we will get a combination of all the changes described above (figure 9 bottom).

Fig. 8: APSD of the absolute displacement signal - Baseline and with degraded hold down springs of pressure vessel/core barrel

Fig. 9: APSD of the absolute displacement signal - Baseline and with degraded hold down springs of pressure vessel/upper core support

The background for such investigation was that, in 1980, at one 4 loop 1200 MW PWR a degradation of the hold-down springs was detected by vibration monitoring. So measurements at a plant in operation with and without degraded hold-down springs are available for comparison purposes.

The plant modeled for the sensitivity studies and the plant with the defect mentioned are not identical. In spite of this fact, we expected comparable effects, since the structure of the primary system and accordingly the vibrative behaviour of both plants are quite similar.
Measurements and model calculations are shown in figure 10. If a degradation of the rotational spring of the core barrel by 44% and a degradation of the upper core support spring by 20% is assumed, the best coincidence between measurement and calculation is reached.

In detail we found that
- the core/core barrel pendular frequency decreases by 15%
- the pressure vessel/core barrel pendular frequency decreases by 5%
- the amplitude of the pressure vessel vertical resonance increases by a factor of 1.5 to 2 without frequency change.

On-situ tests of the hold-down springs during the refueling phase showed that about 50% of the 112 springs were significantly below the specified spring force, 32% even below the lower limit allowed.

**Fig. 10:** APSD of the absolut displacement signal with degraded hold down springs of pressure vessel
SUMMARY

In general, the results of the sensitivity studies described here and of numerous similar investigations performed by means of the 4 loop analytical model can be summarized as follows:

- The monitored PSD functions of the structural vibrations are very sensitive to mechanical failures.
- Each kind of investigated failure causes its own pattern of spectral changes.
- Each pattern of spectral changes is characteristic of one particular failure.
- The failure-caused changes can be observed in more than one VMS-signal. However, one most sensitive spectrum exists for each kind of failure, i.e. this spectrum shows the most serious changes or it shows an incipient failure in the earliest stage of development.
- It is possible to simulate detected changes of the monitored PSD functions using an analytical model. Thus the reason for changes can be identified and adequate countermeasures can be taken.

All these results are encouraging for an extended use of Vibration Monitoring at primary systems of Nuclear Power stations.

REFERENCES


/4/ Sunder R., Wach D., Reactor Diagnosis Using Vibration and Noise Analysis in PWRs, Symposium on Operational Safety of Nuclear Power Plants (1983), Marseille

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Table 1: Identification of the resonance frequencies of a 4-loop-PWR primary system
THREE DIMENSIONAL REACTOR NOISE ANALYSIS FOR PREDICTION OF A CONTROL ROD POTENTIAL FAILURE

Mervat A. Shinaishin and Mahmoud Soubhi
Atomic Energy Authority, Cairo, Egypt

Abstract—Attention is paid in this work to the diagnostics of control rod vibration in power reactors. The diagnosis is based on the analysis of noise signals available from ex-core period meter and neutron detectors. The constraining effect of the guiding tube on the rod vibrational movement is considered in modeling the phenomenon. Rod vibration is considered to be stochastic and two dimensional. Two-energy group diffusion Equations are solved for the neutron fluxes and the adjoint fluxes in the core and the reflector. Localization of the malfunctioning rod is achieved via a deduced correlation. An expression is also obtained for estimating the mean square of the vibration displacement. An application is made on a typical PWR.

1. INTRODUCTION

The interest in early failure detection and diagnosis in Nuclear Power plants (NPPs) is increasing worldwide [1-7, 9-11] especially as a consequence of the successful use on noise diagnosis during the post TMI-accident for prediction of failing sensors. Thus, more and more online monitoring and diagnostic techniques are applied especially in those radioactive inaccessible parts of the Reactor coolant pressure boundary (RCPB). Such online capability will undoubtedly help to enhance reactor safety and availability by minimizing radiation exposure to repair personnel and reducing unforeseen reactor shutdowns.

Among those areas of interest mechanical vibration of control rods is one of particular importance as they are prone to malfunctions because of their active movement. Rod vibration causes reactivity perturbation which in turn leads to neutron flux noise signals. Such signals can be online monitored and analyzed. However, realization of an efficient diagnosis system is faced, up till now, with the difficulty that interpretation of signal spectra or correlation functions is lacking completeness and certainty to the extent that even in installed systems there still hesitation from plant operators to make efficient use of it.

As a consequence, the effort paid in this work is directed towards this area of signal diagnosis and interpretation. In this respect, it is due to mention that in spite of the fact that in-core sensors are not part of the reactor safety systems, and that in-core measurements up till now are not part of the standard vibration Monitoring systems (VMS) installed in new plants all over the world, the analysis made in published literature [2, 4, 7] are based on in-core neutron detector measurements. Interpretation of these measurements is difficult due to the strong influence of local effects. Thus diagnosis in this work is based on noise signals available from ex-core reactor period Meter and ex-core power level neutron detectors. Also in the analysis the constraining effect of the guiding tube on the rod vibrational movement is considered in modeling the phenomenon. Rod vibration is taken to be stochastic and two dimensional. The effect of such vibration of one partially withdrawn (or totally inserted) Rod Cluster Control Assembly (RCCA) on the neutron flux and core reactivity is modeled. In doing so only one element of the cluster is assumed to vibrate. Two energy group diffusion equations are solved for the neutron fluxes and the adjoint fluxes in the core and the reflector.
The objectives of such diagnosis is the identification of the vibrating rod location, and estimation of the vibration mean size. These have been achieved numerically, and application is made on a typical pressurized water Reactor (PWR).

2. MODEL DESCRIPTION

2.1 Neutron flux

Fast and thermal neutron flux distributions in the core are obtained by solving the following two-energy group diffusion equations in the cylindrical coordinates \((r, \theta, z)\):

\[
D_1 \nabla \cdot \nabla \phi_1 - \left( \Sigma_{al1} + \Sigma_{rl1} + \Sigma_{1c} \right) \phi_1 + \Sigma_{f1} \phi_1 + \Sigma_{2c} \phi_2 = 0 \tag{1}
\]

\[
D_2 \nabla \cdot \nabla \phi_2 - \Sigma_{a2c} \phi_2 + \Sigma_{r2c} \phi_2 = 0 \tag{2}
\]

and in the reflector:

\[
D_1 \nabla \cdot \nabla \phi_1 - \Sigma_{r1r} \phi_1 = 0 \tag{3}
\]

\[
D_2 \nabla \cdot \nabla \phi_2 - \Sigma_{a2r} \phi_2 = 0 \tag{4}
\]

These equations are solved under the continuity conditions of both the fast and thermal fluxes and neutron currents at core/reflectors interfaces. It is also required that both reflector fluxes vanish at the extrapolation (radial and axial) distances of the reflector. In addition to these boundary conditions which should be satisfied in a homogenized system, the following conditions are assumed to be satisfied at the vibrating control rod boundary surface:

\[
\frac{\partial \phi_1}{\partial r} = 0 \tag{5}
\]

\[
\frac{\partial \phi_2}{\partial r} = \frac{\phi_2}{d_{cr}} \tag{6}
\]

where \(d_c\) and \(d_{acr}\) are the control rod extrapolation distance and absorption cross section, and conventional notation is used for the parameters and variables appearing in Eq. (1) through (6).

2.2. Adjoint Flux

The adjoint flux equations in the core are

\[
D_1 \nabla \cdot \nabla \psi_1 - \left( \Sigma_{al1} + \Sigma_{rl1} + \Sigma_{f1} \right) \psi_1 + \Sigma_{2c} \phi_2 = 0 \tag{7}
\]

\[
D_2 \nabla \cdot \nabla \psi_2 - \Sigma_{a2c} \psi_2 + \Sigma_{r2c} \psi_2 = 0 \tag{8}
\]

and in the reflector:

\[
D_1 \nabla \cdot \nabla \psi_1 - \Sigma_{r1r} \psi_1 = 0 \tag{9}
\]

\[
D_2 \nabla \cdot \nabla \psi_2 - \Sigma_{a2r} \psi_2 = 0 \tag{10}
\]

where \(\psi\) is required to satisfy the same boundary conditions as \(\phi\) in a homogenized core and reflector. In addition, boundary conditions at the assumed thermal neutron flux detector location implies a discontinuity in \(\frac{\partial \phi_2}{\partial r}\) at that location.

2.3. Vibration induced Reactivity

Changes in core reactivity due to a control rod vibration is basically arising from changes in the effective absorption cross section of thermal neutrons in the core. Thus

\[
\Sigma_{a2c} = \Sigma_{acr} + \delta \Sigma_{a2c} \tag{11}
\]

where subscript \(v\) means with vibration; and \(\delta\) is a perturbation symbol.

Let:

\[
3 \Sigma_{a2c} = \Sigma_{acr} \ast (\gamma - (E_{av} + \varepsilon)) \ast (\theta - \frac{\varepsilon}{R_v}) \tag{12}
\]

where \(\Sigma_{a2c}\) is the control rod absorption cross section; \(R_v\) is the radial distance from the core center to a point at the vibrating control rod surface;
Making some simplifications we get the following localization equation at the resonance frequency $\omega_0$ of the vibrating control rod:

$$v\psi(R_v, R_d, \theta_d) - \left\{ \frac{\psi(R_v, R_d, \theta_d)}{\psi(R_v, R_d, \theta_d)} \psi(R_v, R_d, \theta_d) \right\}_w =$$

$$+ \left\{ \frac{\psi(R_v, R_d, \theta_d)}{d(c)} \right\} \left[ \text{APSD} \left( \theta_d \right) + \text{APSD} \left( \theta_d \right) + \text{CPSD} \left( \theta_d \right) \right]$$

$$w = \omega_0$$

(19)

The R.H.S. of this equation is a known constant quantity since it depends on measurable quantities except $d(c)$ which can easily be calculated. The L.H.S. is a function of $R_v$ since $R_d, \theta$ are known by selecting two specific detectors. Thus by plotting the two sides of Eq.(19) as a function of $R_v$, we obtain a curve for the R.H.S. which shall intersect the horizontal straight line representing the R.H.S. at the vibrating rod location.

2.6 Mean square of the vibration displacement

Once $R_v$ is obtained using Eq.(19), the mean square value of the vibration displacement is obtained from

$$\langle \xi^2 \rangle = \frac{1}{A(R_v)} \langle \xi^2 \rangle$$

(20)

where

$$A(R_v) = \frac{d(c) - d(v)}{d(c) - d(v)} \frac{\psi_v(R_v)}{\psi_v} \left[ \begin{array}{c} \frac{1}{d(c)} \\ \frac{1}{d(v)} \end{array} \right]$$

(21)

3. APPLICATION

In our application we selected a PWR which has the following data: core thermal power 3411 MWt, active core equivalent diameter 168.53 cm; core height 364 cm; reflector thickness 25.4 cm on top and bottom, and 38.1 cm on the sides; control Rod(CR) material Ag–In–Cd 80%–15%–5% CR outer diameter $d_v$ 0.8585 cm; guiding tube inner diameter $d_t$ 1.143 cm; $\sigma 2.43$; $L_{20} 0.0762 cm$; $L_{10} 0.012 cm$; $L_{20} 0.014 cm$; $L_{10} 0.1456 cm$; $D_{1.263 cm}$; $D_{2} 0.354 cm$; $D_{21} 1.13 cm$; $D_{22} 0.16 cm$; $D_{23} 0.042 cm$; $D_{24} 0.0197 cm$; $L_{21} 1.078 cm$; and $L_{22} 0.32 cm$.

For the R.H.S. of Eq.(19) to exist we have to assume an initiating perturbation $\xi(\omega)$, which is typically $2, 4, 7$ expressed in terms of the APFD $\xi(\omega)$ as follows:

$$\text{APFD} \xi(\omega) = \frac{2\langle \xi^2 \rangle}{{(1 + \omega^2 - \sqrt{1 + \omega^2})}}$$

(22)

Values for the damping factor $\gamma$ and the resonance frequency $\omega_0$ of a typical PWR control element are 0.15 and 3 Hz, leading to:

$$\text{APFD} \left[ \xi(\omega) \right] = \frac{1.83 \langle \xi^2 \rangle}{{(\omega^2 - \gamma^2) + 1.82}}$$

(23)

$$\text{APSD} \left[ \xi(\omega) \right] = \frac{1.05 \langle \xi^2 \rangle}{(\omega^2 - \gamma^2) + 1.82}$$

(24)

$$\text{APSD} \left[ \xi(\omega) \right] = \frac{0.05 \langle \xi^2 \rangle}{(\omega^2 - \gamma^2) + 1.82}$$

(25)

i.e.

$$\text{APSD} \left[ \xi(\omega) \right] = 0.04413 \langle \xi^2 \rangle$$

(26)

The ex-core detectors are assumed to be located at a radial distance $R_d$ of 180 cm, and angles $\theta_d$'s 45°, 135°, 225°, 315°.
and ε is the displacement of the rod vibrating around its equilibrium position. Induced reactivity is obtained from:

$$\delta f_v = \frac{\phi_{2c}^\text{core} \phi_{2c}^\text{d} \phi_{2c}^d}{\int_{V_0} V_{\text{core}} e_{ac} \, dv}$$  \hspace{1cm} (13)

Performing these integrations, and noting that the rod is not a wire placed at its center line but rather has a diameter d_v; taking the constraining effect of the rod guiding tube having diameter d_t on the vibration into account; and making use of the relation:

$$\frac{d\phi_{2c}}{dr} \bigg|_{r=R_v} \epsilon(t) = \frac{\phi_{2c}(R_v)}{d_v} \epsilon(t)$$ \hspace{1cm} (14)

we get

$$\delta f_v = -\frac{\Sigma_{\text{ac}} (d_t - d_v)^2 L_v}{v_{f2}} \frac{\phi_{2c}(R_v)}{\phi_{2c}^{\text{core}}} \left[ \frac{\epsilon(t)}{d_v} + \frac{\epsilon(t)}{R_v} \right]$$ \hspace{1cm} (15)

where L_v is the inserted length of the vibrating rod; and \( \overline{\phi}_{2c} \) is the average thermal flux in the core.

Eq. (15) has the same form in the frequency domain; i.e.,

$$\delta f_v(R_v, L_v, \omega) = -\frac{\Sigma_{\text{ac}} (d_t - d_v)^2 L_v}{v_{f2} A_{\text{core}}} \frac{\phi_{2c}(R_v)}{\phi_{2c}^{\text{core}}} \left[ \frac{\epsilon(\omega)}{d_v} + \frac{\epsilon(\omega)}{R_v} \right]$$ \hspace{1cm} (16)

where the rod is assumed to be fully inserted, and \( A_{\text{core}} \) is the cross-sectional area of the core.

2.4. Flux Perturbation

Since the ex-core neutron detectors practically detect thermal neutrons, the thermal neutron flux perturbation due to control rod vibration is

$$\delta \phi_2(R_d, \theta_d, t) = \frac{f_{v \text{core}} \phi_{2c}^d \phi_{2c}^d \phi_{2c}^d}{f_{v \text{cor}} \psi_{f2}}$$ \hspace{1cm} (17)

where \( (R_d, \theta_d) \) refers to a detector location, and \( \psi \) has the same dimension as \( \phi \).

Going through derivations similar to those made in Sec. (2.3), normalizing the perturbation relative to the nominal value of the flux \( \phi(R_d) \) at a detector location; and making Fourier transformation of the resulting equation we get:

$$\delta \phi(R_d, \theta_d, \omega) = -\frac{\Sigma_{\text{ac}} (d_t - d_v)^2}{v_{f2} A_{\text{core}}} \frac{\phi_{2c}(R_v)}{\phi(R_d)} \left[ \frac{\epsilon(\omega)}{d_v} + \frac{\epsilon(\omega)}{R_v} \right] \psi(r, \theta, R_d)$$ \hspace{1cm} (18)

where

$$\frac{\delta \phi}{f_v} = \frac{\delta \phi}{f_{v \text{cor}}} + \frac{1}{\psi} \frac{\delta \psi}{f_{v \text{cor}}}$$

and \( \psi \) is assumed to lie in the plateau region (0.5 to 50 rad./sec.)/4,7/ of the adjacent flux. This condition is valid around the resonance frequency, about 3 Hz/8,11/, of control rods in typical PWRs.

2.5 Localization of vibrating control rod

In order to get an expression which is solely function of the vibrating rod location, Eq. (18) is solved for \( |\epsilon(\omega)| \) and \( \epsilon(\omega) \), using the noise measurements \( \delta \phi(R_v, R_{d1}, \theta, \omega) \) and \( \delta \phi(R_v, R_{d2}, \theta, \omega) \) as two ex-core neutron detector locations.

At this point it may be worth to remind that there exist four full length (two sections each) of uncompensated ionization chambers known as the power range neutron detectors which are positioned in instrument wells at known locations around the reactor pressure vessel.

From the equations for \( |\epsilon(\omega)| \) and \( \epsilon(\omega) \) we get expressions for the Auto Power Spectral Densities APSD(\( \epsilon(\omega) \)) and APSD(\( |\epsilon(\omega)| \)) in terms of the APSD(\( \delta \phi_{d1} \)), APSD(\( \delta \phi_{d2} \)) and the Cross Power Spectral Density CPSD(\( \delta \phi_{d1}, \delta \phi_{d2} \)) of the normalized noise signals. Then from Eq. (16) the APSD(\( \delta f_v \)) is obtained in terms of the spectral densities of the flux noise signals at the two chosen detectors.
Since physical measurements for the quantities on the R.H.S. of Eq. (19) depend themselves on the location \( R_y \) of the vibrating control rod, we have to generate a so-called numerical experiment by assuming a value for \( R_y \) to be used on the R.H.S of Eq. (19) in order to achieve a situation which resembles a physical one. Then the L.H.S. of Eq. (19) is plotted as a function of \( r \). The correlation is validated as shown in Fig. 1 for a vibrating control rod at the center line of the core.

FIG. I Localization of a Vibrating CR

4. CONCLUSION

In this work a means has been developed for localizing a vibrating control rod, and estimating the mean size of the vibration displacement. It has the advantage of being straightforward where we avoided the approach of generating multiple contours as it is found in the published literature, and is thus easier for the operators to comprehend. It, in addition, accounts for the physical conditions arising from the presence of the guiding tube which constrains the maximum displacement of the rod and does not leave it free as has been previously done. The analysis is based on the use of ex-core measurements since as a rule in-core measurements are not part of the safety system. Finally, the developed work has been numerically verified using the relevant data for a typical PWR.

REFERENCES

ANALYSIS OF IN-CORE DYNAMICS IN PRESSURIZED WATER REACTORS WITH APPLICATION TO PARAMETER MONITORING

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ABSTRACT - The behavior of the phase relationship between neutron flux and core-exit temperature fluctuations in a pressurized water reactor (PWR) is studied as a function of the moderator temperature coefficient of reactivity ($\sigma_T$). PWR operational data indicates that the neutron noise and core-exit temperature noise cross power spectrum phase is linear in a certain frequency range, and approaches -180 deg at low frequencies. Extensive modeling studies applied to the LOFT reactor shows that this low frequency phase behavior changes when $\sigma_T$ is positive, approaching zero deg at low frequencies. The analysis further showed that in the LOFT reactor, coolant flow rate fluctuation is the primary driving source causing neutron noise and core-exit temperature fluctuations. This conclusion was also confirmed by independent studies. The neutron noise-coolant temperature phase behavior may be used as a simple method of monitoring the moderator temperature coefficient of reactivity during different stages of a PWR fuel cycle.

1. INTRODUCTION

In-core dynamics of fluctuations in process variables in pressurized water reactors (PWRs) was studied in the past by several investigators (Katona et al., 1982; Sweeney et al., 1985; Glockler et al., 1986), with primary applications to flow monitoring, and detecting the effects of reactivity changes on neutron noise spectrum. Recent research in this area (Shieh, 1985, Shieh et al., 1987) revealed the connection between neutron noise and core-exit coolant temperature noise cross power spectral density (CPSD) phase behavior as a function of the moderator temperature coefficient of reactivity ($\sigma_T$) in a PWR. Some vendors require that $\sigma_T$ be nonpositive for nominal power operation (1973, 1974, 1975). The direct method of $\sigma_T$ measurement requires corrections (to remove fuel temperature feedback effect) based on experimental and theoretical calculations. The complexity of establishing the value of this parameter motivated us to develop a simple but effective method of monitoring the sign of $\sigma_T$ using PWR operational data.

The paper presents a systematic modeling of in-core noise signal dynamics in a PWR with point reactor kinetics behavior, such as in the Loss-of-Fluid Test (LOFT) reactor. The multidimensional model analysis showed (Shieh, 1985) that the feasibility of using the noise signals to monitor $\sigma_T$ depends on the perturbation source. The results of modeling analysis of the LOFT reactor showed that (a) there is a range of frequencies in which the phase between the in-core neutron detector noise and core-exit temperature noise is linear, (b) the frequency range of the linear phase behavior is limited by the primary sink frequency of the corresponding transfer function, (c) for the case when $\sigma_T$ is negative, the phase angle at low frequencies approaches -180 deg, (d) the phase angle approaches zero-degree when $\sigma_T$ is positive, and (e) from the results of the model study and from PWR operational data, it can be concluded that in the LOFT reactor the primary perturbation source is the cool end flow rate fluctuation. This last conclusion was recently established by a multivariate autoregressive analysis of pump $\Delta p$, core $\Delta p$, in-core neutron detector and core-exit thermocouple signal analysis (Glockler and Upadhyaya, 1987), and by experimental measurements at LOFT and commercial PWRs (Sweeney et al., 1985).

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In Sect. 2 the results of in-core data analysis from four different PWRs are summarized. A detailed nodal model of PWR neutronic-thermal hydraulic dynamics and the computation of various frequency domain signatures from the model are described in Sect. 3. The results of model simulation and comparison with experimental data analysis are discussed in Sect. 4. Summary and concluding remarks are given in Sect. 5.

2. RESULTS OF PWR IN-CORE SIGNAL ANALYSIS

In order to establish the phase behavior between the neutron noise signal and the core-exit thermocouple signal at low frequencies we will present results of data analysis from different PWRs. We want to emphasize that the higher frequency behavior of calculated phase will be affected by the location of the detectors (in-core, ex-core neutron detectors and the location of thermocouples from the core-exit). Noise data from four different PWRs were analyzed. These plants are

(1) 55 MW (thermal) LOFT pressurized water reactor.
(2) 1140 MW (electric) commercial PWR (USA).
(3) 477 MW (electric) Borssele PWR (Netherlands).
(4) 440 MW (electric) Paks PWR Unit II (Hungary).

Figure 1 shows the CPSD phase relationship between core-exit thermocouple and in-core neutron detector signals in the LOFT reactor at 100% power and 100% coolant flow. The LOFT reactor contains cobalt self-powered neutron detectors (ND) at four different axial positions, 27.9, 68.6, 111.8 and 154.9 cm above the bottom of the core. The core-exit coolant temperature was measured by K-type thermocouples (TC) located at fuel bundle upper grid. The extrapolated linear phase approaches -180 deg at low frequencies. We want to consider the frequency range (above 0.1 Hz) where the flow propagation effect is dominating the phase. Similar behavior was observed at power levels 25%, 50% and 75%, and at different flow rates. As the flow rate decreases, the phase slope increases, indicating larger transit time (Sweeney et al, 1985).

Figure 2 shows the neutron-temperature CPSD phase relationship in a 1140 MW (electric) commercial PWR at 100% power. The phase has a linear behavior in the frequency range 0.1-1.5 Hz, with the low frequency extrapolation approaching -180 deg. Figure 3a shows the phase between in-core ND and core-exit TC in the Borssele PWR (477 MW) at full power and flow condition. The corresponding CPSD is shown in Fig. 3b which shows a significant sink frequency at about 2.35 Hz (Upadhyaya and Turkcan, 1984). Sink frequencies are characteristics of systems with flow and heat transfer (Kosaly and Hesko, 1972). The sink frequency is a function of the flow rate and the axial flux shape (Shieh, 1985). Finally, the phase between in-core ND and core-exit TC signals for the Paks Unit II PWR is shown in Fig. 4 (the phase is linear in the frequency range 0.1-0.8 Hz). Once again the out of phase behavior at low frequencies is evident.

In all the above cases the moderator temperature coefficient of reactivity was known to be negative. The extrapolated phase angle approaches -180 deg in all the cases. These are strong evidences for us to postulate that when \( q_c \) is positive the extrapolated phase angle at zero frequency would approach the zero-deg phase. This method is shown to be "not sensitive to changes in the Doppler coefficient and provides a nonambiguous technique for monitoring the sign of \( q_c \)."

3. DYNAMIC MODELING OF PROCESS FLUCTUATIONS IN A PWR CORE

3.1. Purpose of Modeling

The primary goal of modeling discussed in this section is to study the dynamics between neutron power and coolant temperature, and the effects of varying \( q_c \). The dynamic behavior depends on the perturbation source, and it is necessary to establish this cause for a given PWR. The fuel temperature and the moderator temperature affect the neutron power in different frequency ranges. Because of large fuel to coolant heat transfer time constant, the higher frequency effects (above 0.1 Hz) on the coolant temperature fluctuations will be attenuated. Conversely, the fuel temperature fluctuations will not be affected due to high-frequency fluctuations in the moderator temperature. Thus, in the frequency range of interest to us (above 0.1 Hz), the fuel temperature coefficient of reactivity is not relevant, and the higher frequency moderator temperature-neutron power dynamics is studied. Thus the identification of noise perturbation source and the choice of proper frequency range is important in monitoring the \( q_c \) parameter in a PWR.

A multinodal model of the LOFT reactor was developed to study neutronics-thermal hydraulics dynamics. This consists of point reactor kinetics with two delayed-neutron groups, 10 fuel nodes, 10 cladding nodes, and 20 coolant nodes. The perturbation sources considered are: core inlet temperature disturbance, core coolant velocity disturbance, random heat transfer
disturbance, and heat source perturbation. For the coolant velocity disturbance we consider simultaneous changes in the velocity along the channel, and possibly caused by fluctuations in pump Δp. The random heat transfer disturbance is assumed to be caused by turbulent flow, causing random changes in the local heat transfer coefficient.

3.2. The Nodal Model Development

The multinodal model of the LOFT reactor as described by Shieh et al (1985) is given below. A single channel geometry with a cylindrical fuel region is considered with n fuel nodes, n cladding nodes, and 2n coolant nodes. (See Table 1 for Nomenclature).

Neutronic equations:

\[
\frac{d}{dt} \left( \frac{\delta N}{N_0} \right) = - \frac{\beta}{\lambda} \frac{\delta N}{N_0} + \frac{a_f}{\lambda} \frac{1}{N_{w1}} \sum_{i=1}^{n} T_{fi} W_{fi} \left[ \lambda_i \frac{\delta C_i}{N_0} + \frac{S(t)}{A} \right] + \frac{a_c}{\lambda} \frac{1}{W_{ci}} \sum_{i=1}^{2n} T_{ci} W_{ci} \left[ \lambda_i \frac{\delta C_i}{N_0} \right].
\]

(1)

Fuel temperature, node i=1,2,..., n:

\[
\frac{d}{dt} \frac{\delta T_{fi}}{N_0} = \frac{\beta_f}{\lambda_f} \frac{\delta N}{N_0} - \frac{\lambda_i \delta C_i}{W_{ci}} - \frac{h_{fc1} A_{fc1}}{m_{ci} c_{pci}} \left( \delta T_{fi} - \delta T_{ci} \right) .
\]

(3)

Cladding temperature, node i=1,2,..., n:

\[
\frac{d}{dt} \frac{\delta T_{ci}}{m_{ci} c_{pci}} = \frac{h_{fc1} A_{fc1}}{m_{ci} c_{pci}} \left( \delta T_{fi} - \delta T_{ci} \right) - \frac{h_{cl1} A_{cl1}}{m_{cl1} c_{cl1}} \left( \delta T_{ci} - \delta T_{(2i-1)} \right) .
\]

(4)

Coolant node 1:

\[
\frac{d}{dt} \frac{\delta T_{c1}}{m_{ci} c_{pci}} = \frac{\beta_f}{\lambda_f} \frac{\delta N}{N_0} \frac{F_{ci}}{m_{ci} c_{pci}} \left[ \delta T_{ci} - \delta T_{c1} \right] + \frac{m_{cl1}}{m_{ci}} \left( T_{in} - T_{ci} \right).
\]

(5)

Coolant nodes i=2,4,..., 2n:

\[
\frac{d}{dt} \frac{\delta T_{ci}}{m_{ci} c_{pci}} = \frac{\beta_f}{\lambda_f} \frac{\delta N}{N_0} \frac{F_{ci}}{m_{ci} c_{pci}} \left[ \delta T_{ci} - \delta T_{c1} \right] + \frac{h_{clc(i/2)} A_{clc(i/2)}}{2m_{clc} c_{pci}} \left( \delta T_{ci} - \delta T_{(i-1)} \right) + \frac{m_{clc(i/2)}}{m_{clc} c_{pci}} \left( \delta T_{c(i-1)} - \delta T_{ci} \right)
\]

(6)

Coolant nodes i=3, 5, ..., 2n-1:

\[
\frac{d}{dt} \frac{\delta T_{ci}}{m_{ci} c_{pci}} = \frac{\beta_f}{\lambda_f} \frac{\delta N}{N_0} \frac{F_{ci}}{m_{ci} c_{pci}} \left[ \delta T_{ci} - \delta T_{c1} \right] + \frac{h_{clc(i+1)} A_{clc(i+1)}}{2m_{clc} c_{pci}} \left( \delta T_{ci} - \delta T_{c1} \right) + \frac{m_{clc(i+1)}}{m_{clc} c_{pci}} \left( \delta T_{c(i-1)} - \delta T_{ci} \right)
\]

(7)
Thermocouple output:

\[
\frac{d}{dt} \delta T_a = \frac{\delta T_{c2} x_0}{t} - \frac{\delta T_m}{t}
\]  

(8)

If the coolant flow rate changes, the heat transfer coefficient will also change. Modeling this by the Dittus-Boelter equation (Re>2000) for the Nusselt number (El-Wakil, 1978)

\[
N_u = \frac{hD}{k} = 0.023(Re)^{0.8} (Pr)^{0.4}
\]  

(9)

Reynolds number, \( Re = \frac{Dvd}{\nu} \)

(10)

and Prandtl number, \( Pr = \frac{C_v}{k} \)

(11)

where \( D \) = equivalent diameter of coolant channel, \( f \) = coolant density, \( \nu \) = viscosity, \( h \) = heat transfer coefficient, \( c \) = specific heat capacity of coolant, \( k \) = coolant thermal conductivity. Using Eq. (9), the fluctuations in \( h \) and \( \nu \) are related by

\[
\frac{\delta h}{h} = 0.8 \frac{\delta \nu}{\nu}
\]  

(12)

The cladding/coolant interaction is then given by

\[
\delta h_{c1c1} = 0.8 h_{c1c1} \frac{\delta m}{m}
\]  

(13)

The fluctuations in the coolant mass flow rate is factored into Eqs. (4)-(7). (See Shieh et al, 1985.) The weighting of the i-th node fuel temperature on the reactivity, \( W_{f1} \), and the weighting of the i-th node coolant temperature on the reactivity, \( W_{c1} \), are proportional to the square of the fraction of the power deposited in the i-th node, namely, \( F_{f1} \) and \( F_{c1} \).

The above set of equations describe a linear dynamics

\[
\dot{X}(t) = AX(t) + BU(t)
\]  

(14)

where \( X \) = vector of state variables, \( U \) = input vector, \( A \) = system matrix, and \( B \) = input matrix.

3.3. Calculations in the Frequency Domain

Equation (14) is transformed to the frequency domain and is solved for the Fourier transform of \( X(t) \),

\[
X(\omega) = (j\omega A)^{-1} BU(\omega)
\]  

(15)

where \( j = \sqrt{-1} \), \( I \) = identity matrix, \( \omega \) = frequency. The transfer function between the state variable \( x_i \) and the input variable \( u_k \) is given by the matrix element \( [(j\omega A)^{-1} B]_{ik} \). The transfer function between the state variables \( x_i \) and \( x_j \) with input \( u_k \) has the form

\[
G_{ij}(\omega) = \frac{[(j\omega A)^{-1} B]_{ik}}{[(j\omega A)^{-1} B]_{jk}}
\]  

(16)

Thus from the matrix \( [(j\omega A)^{-1} B] \) we can obtain transfer function between state variables (say \( \delta T_{out} \) and \( \delta N \)) for any specified input (such as coolant velocity perturbation and inlet coolant temperature perturbation).
4. APPLICATION TO THE LOFT REACTOR

4.1. Frequency Domain Results

We calculated phase and transfer function between core-exit coolant temperature and neutron power fluctuations for the inputs stated above. LOFT reactor design parameters (Reeder, 1978) were used to complete the model description. The phase relationship is linear in a certain frequency range and its slope is inversely proportional to the coolant velocity. The maximum linear phase frequency is limited by the primary sink frequency, which in turn depends on the coolant velocity and the axial flux shape. The sink frequency is a frequency at which the net fluctuation in a given signal goes to zero (Kosaly and Mesko, 1972; Kosaly et al., 1982). For the LOFT reactor, under normal conditions the linear phase frequency range is about 0.1-2 Hz (see Fig. 5) and the sink frequency is about 2.2 Hz (see Fig. 6). In Fig. 6, the transfer function gain, with inlet temperature fluctuation as the perturbation source has a peak instead of a sink effect. The primary sink frequency can be generally expressed as (Shieh, 1985)

\[ f_s = \frac{c \nu}{L} \] (17)

where \( \nu \) = coolant velocity (m/s), \( L \) = active core length (m), \( c \) = constant in the range, 1<\( c <2. \)

A very important observation from these results is that the extrapolated phase approaches \(-180\) deg. at low frequencies. The nominal value of \( \alpha_c \) is \(-0.07675/C\). When \( \alpha_c \) is positive \((>0.0035/C)\), the low frequency phase behavior changes, with the extrapolated phase approaching zero-deg at low frequencies (see Fig. 7). The analysis clearly shows that the extrapolated neutron - core-exit temperature phase at low frequencies \((-180\) deg for negative \( \alpha_c \), zero-deg for positive \( \alpha_c \)) depends on the sign of the moderator temperature coefficient of reactivity. We do not attempt to explain the phase behavior below about 0.1 Hz because of the combination of various effects in the 0-0.1 Hz frequency region. The results also show that if the heat source perturbation (which affects reactor power first and through heat transfer affects the coolant temperature) is the dominant source, the low frequency phase behavior is not affected by the sign of \( \alpha_c \). The extrapolated phase will not approach \(-180\) deg. at low frequencies.

4.2. Perturbation Source in the LOFT Reactor Core

Our analysis also indicated that the dominant perturbation source giving rise to neutron and core-exit temperature fluctuations is the core coolant flow rate fluctuations. The following evidence is presented to prove this fact related to the LOFT reactor.

1. Very high coherence (\(=0.9\)) was observed by Canon and Clemo (1980) between neutron noise and coolant flow noise as measured by a venturi meter.
2. The above observation excludes the random heat transfer as the dominant disturbance.
3. Direct reactivity induced perturbations are excluded because of the entirely different nature of the phase behavior (Shieh et al., 1987).
4. The low coherence between inlet coolant temperature \( T_{in} \) and core-exit temperature, and \( T_{in} \) and neutron noise, excludes the possibility that inlet coolant temperature is the perturbation source (Upadhyaya, 1982).
5. Independent multivariate signal analysis of the LOFT core subsystem (Glockler and Upadhyaya, 1987) showed very clearly that the core coolant flow fluctuation as represented by the primary pump \( \Delta P \) was the driving noise in this system.
6. The model studies also indicate (Shieh, 1985) that if the random heat transfer is the dominating source, then the coherence between core-exit \( TC \) and in-core \( ND \) must be small. On the contrary this value is about 0.6 in the LOFT reactor.
7. As discussed in Sect. 4.1, the comparison of the model analysis result and the experimental result excludes the heat source perturbation as the dominant source in the LOFT reactor.

5. SUMMARY AND CONCLUDING REMARKS

We presented a detailed theoretical analysis of the in-core dynamics and perturbation sources affecting neutron and core-exit temperature noise relationship. Even though this analysis was developed for a point reactor kinetics system, the experimental results from PWRs of different sizes indicates that the moderator temperature coefficient of reactivity influences the behavior of CPFD phase between neutron noise and core-exit temperature noise. The following results highlight the modeling analysis of the LOFT reactor in-core dynamics.

1. There is a range of frequencies in which the in-core \( ND \) noise and core-exit \( TC \) signals is linear and that the slope is inversely proportional to the coolant flow rate.
2. The extent of the linear phase behavior is limited by the primary sink frequency of the core-exit coolant temperature and neutron flux transfer functions, and by feedback effects at low frequencies (below 0.1 Hz).

3. For the case when $\alpha_c$ is negative, the extrapolated phase angle at low frequencies approaches $-180$ deg for all disturbance sources except for the direct reactivity perturbations.

4. The above phase angle approaches zero-deg when $\alpha_c$ is positive.

5. The results of the model study and FWR operational data indicate, that in the LOFT reactor, the primary disturbance source causing core-exit coolant temperature and neutron power fluctuation is the core coolant flow rate fluctuation.

This approach for monitoring the reactivity parameter has the following features: (1) The implementation of the method will not interfere with normal plant operation. (2) $\alpha_c$ can be monitored at all operating conditions above 25% full power (this is the range of data available to us). (3) It is easy to track the parameter by plant personnel because of the easy distinction between the phase angles $-180$ deg (for negative $\alpha_c$) and zero-deg (for positive $\alpha_c$). This monitoring scheme is not sensitive to the actual magnitude of $\alpha_c$ or the value of Doppler coefficient, and is "robust with respect to measurement uncertainties."

REFERENCES


<table>
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<th>Symbol</th>
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<tbody>
<tr>
<td>( \delta N )</td>
<td>variation of the neutron power</td>
</tr>
<tr>
<td>( N_0 )</td>
<td>initial neutron power</td>
</tr>
<tr>
<td>( F_{fi} )</td>
<td>fraction of power deposited in the ( i )'th fuel node</td>
</tr>
<tr>
<td>( W_{fi} )</td>
<td>reactivity weighting for the ( i )'th node fuel temperature</td>
</tr>
<tr>
<td>( S(t) )</td>
<td>reactivity disturbance</td>
</tr>
<tr>
<td>( m_{fi} )</td>
<td>( i )'th node fuel mass</td>
</tr>
<tr>
<td>( C_{pfi} )</td>
<td>( i )'th node fuel specific heat capacity</td>
</tr>
<tr>
<td>( T_{fi} )</td>
<td>( i )'th node fuel temperature</td>
</tr>
<tr>
<td>( h_{fcli} )</td>
<td>( i )'th node fuel to cladding heat transfer coefficient</td>
</tr>
<tr>
<td>( A_{fcli} )</td>
<td>( i )'th node fuel to cladding heat transfer area</td>
</tr>
<tr>
<td>( m_{cili} )</td>
<td>( i )'th node cladding mass</td>
</tr>
<tr>
<td>( C_{pclli} )</td>
<td>( i )'th node cladding specific heat capacity</td>
</tr>
<tr>
<td>( h_{clici} )</td>
<td>( i )'th node cladding to coolant heat transfer coefficient</td>
</tr>
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<td>( A_{clici} )</td>
<td>( i )'th node cladding to coolant heat transfer area</td>
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<tr>
<td>( T_{ci} )</td>
<td>( i )'th node coolant temperature</td>
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<tr>
<td>( m_{ci} )</td>
<td>( i )'th node coolant mass</td>
</tr>
<tr>
<td>( C_{pcfi} )</td>
<td>( i )'th node coolant specific heat capacity</td>
</tr>
<tr>
<td>( W_{ci} )</td>
<td>reactivity weighting for the ( i )'th node coolant temperature</td>
</tr>
<tr>
<td>( \dot{m} )</td>
<td>core coolant mass flow rate</td>
</tr>
<tr>
<td>( T_m )</td>
<td>measurement of core-exit coolant temperature</td>
</tr>
<tr>
<td>( \tau )</td>
<td>temperature sensor time constant</td>
</tr>
</tbody>
</table>
Fig. 1. Cross power spectral density phase relationship between in-core neutron detector and core-exit thermocouple noise signals in the LOFT reactor at 100% power and flow condition.

Fig. 2. CPSD phase relationship between ex-vessel neutron detector and core-exit thermocouple noise signals in a 1140 MW(electric) commercial PWR.

Fig. 3a. Phase relationship between in-core neutron detector and core-exit thermocouple signals in the Borssele PWR at full power and flow condition.

Fig. 3b. Cross power spectrum between the signals used in Fig. 3a (Borssele PWR), showing a sink frequency at about 2.35 Hz.

Fig. 4. CPSD Phase relationship between in-core neutron detector and core-exit thermocouple noise signals in the Paks PWR Unit 2.
Fig. 5. The transfer function phase relationship between neutron noise and core-exit temperature noise in the LOFT reactor for various perturbation sources (theoretical model).
(Nucl. Sci. Eng., Vol. 95, 1987)

Fig. 6. The transfer function gain between neutron noise and core-exit temperature noise in the LOFT reactor for various perturbation sources (theoretical model).
(Nucl. Sci. Eng., Vol. 95, 1987)

Fig. 7. The transfer function phase behavior between neutron noise and core-exit temperature noise in the LOFT reactor for various perturbation sources (theoretical model), for the case with positive moderator temperature coefficient of reactivity.
(Nucl. Sci. Eng., Vol 95, 1987)
INVESTIGATION OF BOILING AND PRESSURE FEEDBACK ON NEUTRON NOISE

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Abstract - Boiling and pressure feedback effects on neutron noise in BWR's are investigated with a 1-D reactor model and compared with experimental measurements. It is concluded that the structure of neutron noise coherences at low frequencies can be explained by either pressure or/and boiling feedback.

Keywords - neutron noise, feedback, BWR, boiling, pressure

1. INTRODUCTION

It is well known that at low frequencies and/or small reactor sizes, neutron noise exhibits point behaviour [1]. However as was pointed out by Kleiss and Van Dam [2] boiling feedback reduces the detector field of view, therefore it reduces the coherence of neutron detectors well separated in the core.

However, boiling feedback alone cannot explain the increase of the coherence at even lower frequencies. This increase was attributed [2,3] to feedback from the pressure controller of the reactor. The scope of the present work is to investigate the structure of neutron noise coherences in the low frequency range.

2. THE MODEL

We consider a two region (fuel-coolant), one dimensional reactor model in the radial direction. Let \( x_i \) be the inlet and \( x \) the outlet steam quality at a coolant channel. Energy balance in each coolant channel gives:

\[
\frac{3x}{at} = \frac{\mu H \theta - h_{fg}(x-x_i)}{t_r h_{fg}} \quad (1)
\]

\[
\mu = \frac{\eta_f k}{\rho_c A_c \hat{V}} \quad (2)
\]

where \( h_{fg} \) is the enthalpy of evaporation, \( H \) the channel height, \( \hat{V} \) the coolant velocity, \( t_r = H/\hat{V} \) the coolant transit time, \( \eta_f \) the fuel rod perimeter, \( k \) the heat transfer coefficient (\( W \cdot m^{-2} \cdot K^{-1} \)), \( \rho_c \) the coolant density, \( A_c \) the cross-section area of the coolant channel and \( \theta \) the temperature difference between fuel \( T_u \) and coolant \( T_c \) temperatures.
\[ \theta = T_u - T_c \]  

Consider small random fluctuations \( \delta \theta, \delta x, \delta x_i \) and \( \delta h_{fg} \) of \( \theta, x, x_i \) and \( h_{fg} \) respectively

\[
\begin{align*}
\theta(t) &= \theta_0 + \delta \theta(t) \\
 x(t) &= x_0 + \delta x(t) \\
 x_i(t) &= x_{i0} + \delta x_i(t) \\
 h_{fg}(t) &= x_{fg0} + \delta h_{fg}(t)
\end{align*}
\]  

where the subscript "o" denotes the static value of the respective quantity.

Substituting (4) in (1) and linearizing we get

\[
t_r h_{fg0} \frac{\partial \delta x}{\partial t} = \mu h \delta \theta + h_{fg0}(\delta x_i - \delta x) + \delta h_{fg}(x_{i0} - x_0)
\]  

The enthalpy of evaporation is, by definition,

\[
h_{fg} = h_g - h_f = (p u_g + u_g) - (p u_f + u_f)
\]  

where \( p \) is the pressure, \( u \) the specific volume, \( u \) the specific internal energy, the subscript \( f \) refers to the liquid phase and the subscript \( g \) to the gas phase. Internal energy \( u_f \) and specific volume \( u_f \) of the liquid phase are weak functions of pressure. Therefore the fluctuations \( \delta u_f \) and \( \delta u_f \) induced by pressure fluctuations may be considered negligible compared with the corresponding fluctuations \( \delta u_g \) of the gas phase. Pressure fluctuations, induced by the pressure controller, travel in the channel with the speed of sound, i.e. instantaneously compared with the time required for corresponding temperature fluctuations to develop. We may therefore consider the coolant temperature constant. Treating the steam as a perfect gas we may write

\[
p u_g = R_g T = \text{constant}
\]  

\[
p_0 \delta u_g + u_{g0} \delta p = 0
\]  

where the pressure \( p \) and specific volume \( u_g \) have been separated into static and fluctuating parts.

\[
p(t) = p_0 + \delta p(t)
\]

\[
u_g(t) = u_{g0} + \delta u_g(t)
\]

Using (6), (7), (8) and (9) we get

\[
\delta h_{fg} = -u_{g0} \delta p
\]  

Substitution of (10) into (5) gives
\[
\frac{3\delta x}{\delta t} = c_1 \delta \theta + c_2 \delta p + \delta x_1 - \delta x \tag{11}
\]

\[
c_1 = \frac{\mu H}{h_{fg0}}, \quad c_2 = -u_{f0}(x_{i0} - x_{o0})/h_{fg0} \tag{12}
\]

Replacing \(\delta x\) with \(\delta x = \delta \alpha/\mu_1\) [4], where \(\delta \alpha\) is the void fraction fluctuation and Laplace transforming (11), we get

\[
\delta \tilde{\alpha} = \frac{1}{1 + st_r} \delta \tilde{\alpha}_1 + \frac{\mu_1 c_1}{1 + st_r} \delta \tilde{\theta} + \frac{\mu_1 c_2}{1 + st_r} \delta \tilde{p} \tag{13}
\]

\[
\mu_1 = b^{-1} \alpha(1 - \alpha) e^2, \quad \epsilon = \frac{1}{S} \frac{u_g}{u_f}
\]

where \(S\) is the slip ratio.

The transfer function \(H(j\omega)\) relating pressure fluctuations \(\delta p\) induced by the pressure controller to void fraction fluctuation \(\delta \alpha\), may be written in the form [3].

\[
H(j\omega) = \frac{\delta \tilde{\alpha}(j\omega)}{\delta \tilde{p}(j\omega)} = k_c H_C(j\omega) = k_c \frac{1 + st_{c1}}{1 + st_{c2}} \tag{14}
\]

where the constant \(k_c\) may take values in the range 1 to 11.6 and the characteristic time constants of the pressure controller take values \(\tau_{c1} = 0.9\) to 1.65 sec and \(\tau_{c2} = 3\) to 13 sec [3].

The one-group diffusion equation for the normalized neutron flux stochastic fluctuation reads [5].

\[
\frac{d^2 \tilde{\phi}}{d\xi^2} + b^2 \tilde{\phi} + h^2 \chi \left\{ \frac{Q(j\omega)}{1 + B_g L_o^2} \frac{\nu \delta \tilde{\Sigma}_f}{\Sigma_{ao}} \frac{\delta \tilde{\Sigma}_au}{\Sigma_{ao}} \right\} +
\]

\[
+ \left\{ h^2 \chi \frac{\Sigma_{amo}}{\Sigma_{ao}} - \chi \frac{2n^2 L_o}{L^2} \frac{d^2 \chi}{d\xi^2} \right\} \frac{\delta \tilde{\alpha}}{1 - \alpha} = 0 \tag{15}
\]

\[
\tilde{\phi}(\xi, j\omega) = \delta \phi(\xi, j\omega)/\Phi_0, \quad \xi = x/D, \quad h = D/L_o \tag{16}
\]

\[
\chi(\xi) = \sin \pi \xi, \quad B_g = n/D \tag{17}
\]

\[
b^2 = \pi^2 - h^2 G_0^{-1} \tag{18}
\]

\[
Q(j\omega) = 1 - \frac{\beta \delta}{s + \lambda} \tag{19}
\]

where \(G_0\) is the zero-power reactor transfer function, \(D\) is the reactor size (in the radial direction), \(h\) is the reactor size normalized with the diffusion length \(L_o\), \(L_{so}\) is the slowing down length, \(\Sigma_{amo}\) is the moderator absorption cross-section.
\( \Sigma_{ao} \) is the total absorption cross-section, \( \xi \) is the normalized coordinate, \( \Phi_0 \) is the maximum static neutron flux and the rest of the symbols have their usual meaning with the subscript "o" indicating static values. The fission and fuel absorption cross section fluctuations \( \delta \Sigma_f \) and \( \delta \Sigma_u \) are induced by fuel temperature fluctuations \( \delta T_u \):

\[ \delta \Sigma_u = \nu_u \delta T_u \quad , \quad \delta \Sigma_f = -\nu_f \delta T_u \]  

(20)

Energy balance between fuel and coolant, assuming constant coolant temperature, gives [5]

\[ \delta \tilde{\Phi}(\xi,j\omega) = \frac{P_c}{s+\omega_1 \chi(\xi)+\omega_o} \tilde{\Phi}(\xi,j\omega) \]  

(21)

where \( P_c \) is proportional to the maximum power density, \( \omega^{-1}_0 \) is the relaxation time associated with heat transfer from fuel to coolant and \( \omega^{-1}_1 \) is relaxation time associated with the fuel temperature response to neutron flux changes. From (13), (14), (15), (20) and (21) we finally get

\[ \frac{d^2 \phi}{d \xi^2} + b^2 \phi + P_c h^2 F_1(\xi,s) \left\{ \frac{\mu_1 c_1}{1+st_r-\mu_1 c_2(k_c \hbar c)^{-1}} - v(j\omega) \right\} \phi = -F_3(\xi,j\omega) \delta \tilde{\Phi}(\xi,s) \]  

(22)

\[ F_1(\xi,j\omega) = \frac{\chi(\xi)}{s+\omega_1 \chi(\xi)+\omega_o} \]  

(23)

\[ F_3(\xi,j\omega) = \frac{h^2 \chi(\xi)}{(1-a)(1+st_r-\mu_1 c_2(k_c \hbar c)^{-1})} \times \left\{ \frac{\Sigma_{amo}}{\Sigma_{ao}} - \frac{2\pi^2 L_{so}}{l_o^2 h^2} - \frac{n^2}{h^2} \right\} \]  

(24)

\[ v(j\omega) = Q(\omega) \left( 1 + b^2 L_s^2 \right)^{-1} \frac{\nu_j f}{\Sigma_{ao}} + \frac{\nu_u}{\Sigma_{ao}} \]  

(25)

3. STRUCTURE OF COHERENCES

The system Green’s function \( G(\xi,\xi_0,j\omega) \) reads

\[ \frac{d^2 G}{d \xi^2} + b^2 G + P_c h^2 F_1 \left\{ \frac{\mu_1 c_1}{1+st_r-\mu_1 c_2(k_c \hbar c)^{-1}} - v(j\omega) \right\} G = -F_3(\xi,s) \delta(\xi-\xi_0) \]  

(26)

\[ G(0,\xi_0,j\omega) = G(1,\xi_0,j\omega) = 0 \]  

(27)

The term \( \mu_1 c_2(k_c \hbar c)^{-1} \) represents pressure feedback. Neglecting this term we get the Green’s function in the absence of pressure feedback, whilst boiling feedback is uniform, in the sense that feedback coefficients are constant throughout the core [5]. This case is given by curve 1, in Fig.1.
The Green’s function of eq.(26), i.e. including the pressure feedback term, is given by curve 2 of Fig.1. It can be seen that feedback from the pressure controller, widens the detectors field of view. This is of course to be expected: a local fluctuation of void fraction at \( \xi_0 \), will excite pressure fluctuations through the pressure controller, which will be sensed throughout the core and will induce void fluctuations. These in turn will induce neutron fluctuations. Assuming now the excitation \( \delta a_i \) to be,

\[
\delta a_1(\xi,j\omega) = \delta a_1(j\omega) \delta(\xi-\xi_0) + \delta a_2(j\omega) \delta(\xi-\xi_0) \tag{28}
\]

the neutron response is

\[
\tilde{\phi}(\xi,j\omega) = \int_{0}^{1} \delta a_i(\xi, j\omega) \ G(\xi, \xi_i, j\omega) \ d\xi_0 = \delta a_1(j\omega) G(\xi, \xi_0, j\omega) + \delta a_2(j\omega) G(\xi, \xi_0, j\omega) \tag{29}
\]

If we consider two neutron detectors placed at positions \( \xi_1 \) and \( \xi_2 \), far apart from each other, we may assume that the respective void fraction excitations \( \delta a_1 \) and \( \delta a_2 \) are uncorrelated. Then, the cross-spectrum \( S_{12} \) and the autospectra \( S_{11} \), \( S_{22} \) of the neutronic signals read.

\[
S_{12} = \langle \tilde{\phi}(\xi_1,j\omega) \tilde{\phi}^*(\xi_2,j\omega) \rangle = \\
= |\delta a_1|^2 \ G(\xi_1, \xi_0, j\omega) \ G^*(\xi_2, \xi_0, j\omega) + \\
+ |\delta a_2|^2 \ G(\xi_1, \xi_0, j\omega) \ G^*(\xi_2, \xi_0, j\omega) \tag{30}
\]

\[
S_{11} = \langle \tilde{\phi}(\xi_1,j\omega) \tilde{\phi}^*(\xi_1,j\omega) \rangle = \\
= |\delta a_1|^2 |G(\xi_1, \xi_0, j\omega)|^2 + \\
+ |\delta a_2|^2 |G(\xi_1, \xi_0, j\omega)|^2 \tag{31}
\]
\[ S_{22} = \tilde{\Phi}(\xi_2, j\omega) \cdot \tilde{\Phi}^*(\xi_2, j\omega) = \\
= |\hat{\alpha}_1|^2 |G(\xi_2, \xi_{01}, j\omega)|^2 + \\
+ |\hat{\alpha}_2|^2 |G(\xi_2, \xi_{02}, j\omega)|^2 \] (32)

where the superscript "*" indicates complex conjugate.

The corresponding coherence function,

\[ C_{12} = |S_{12}|^2 / (S_{11} \cdot S_{22}) \] (33)

of the two signals is presented in Fig.2.

![Coherence function between two neutron detectors. Uniform boiling feedback; pressure feedback is included.](image)

This is in agreement with the measured coherences by Kleiss and Van Dam [2]. Similar measurements were reported by Sides et al. [6], where some, but not all, of measured coherences exhibit an increase of the coherence in the very low frequency range similar to that measured by Kleiss and Van Dam [2] and exhibited in Fig.2. If this behaviour was only due to feedback from the pressure controller, then all coherences, between any pair of detectors in the core should exhibit the same behaviour. However, as mentioned already, this is not the case in the measurements by Sides et al. [6]. In fact there are coherences in which this increase is absent even for pairs of detectors placed in the same channel. If the increase in the very low frequency range was due to pressure feedback, and only to that, then one would expect this behaviour to be present at least for detectors in the same channel. We may therefore conclude that there is another mechanism at work. It was shown by Antonopoulos-Domis and Chatziathanasiou [5] that non-uniform boiling feedback may increase or even displace the detector field of view.

Space dependence of feedback may be modelled [5] by the last term of the left hand side of eq.(34).
\[
\frac{d^2 G}{d\xi^2} + b^2 G + \frac{h^2}{c} F_1 \left( \frac{\mu}{1 + s t_r} \right) - \gamma G + f(\xi, j\omega) G = -F_2 \delta(\xi - \xi_0) \tag{34}
\]

\[
F_2(\xi, j\omega) = \frac{h^2 \chi}{(1 - \alpha)(1 + s t_r)} \left\{ \frac{\Sigma_{\text{amo}}}{\Sigma_{\text{ao}}} - \frac{2\pi^2 L_0^2}{h^2 L_0^2} - \frac{\pi^2}{h^2} \right\} \tag{35}
\]

The function \(f(\xi, j\omega)\) represents a space and frequency dependent feedback coefficient. It can be seen that pressure feedback is not included in eq.(34). Assuming the same excitation as in (28) with \(\delta \alpha_1\) and \(\delta \alpha_2\) uncorrelated and boiling feedback non-uniformity to exist only at \(\xi_1\), i.e. assuming

\[
f(\xi, j\omega) = f_1(j\omega) \delta(\xi - \xi_1) \tag{36}
\]

where the function \(f_1(j\omega)\) is given by Fig.3, the resulting coherence between two neutron detectors placed at \(\xi_1\) and \(\xi_2\), is presented in Fig.4. It can be seen that the coherence exhibits the same structure at low frequencies, as that observed by Kleiss and Van-Dam [2], and Sides et al. [6].

\[|f_1|\]

Fig.3. The function \(f_1(j\omega)\)

\[\text{Coherence} \]

\[f(\text{Hz})\]

Fig.4. Coherence without pressure feedback, but with non-uniform (space dependent) boiling feedback.
4. CONCLUSIONS

We may therefore conclude that the structure of coherences of neutron signals at low frequencies, and in particular the increase of some of these coherences at very low frequencies may be due to either pressure feedback or to boiling feedback non-uniformities (space dependence) or both.

REFERENCES

ON THE NEUTRON-NOISE TRANSMISSION STUDIES FOR NON-MULTIPLYING MEDIA USING TRANSPORT THEORY

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Abstract - This paper reports the results of our investigations on the neutron-noise transmission characteristics of non-multiplying media using transport theory. The study has been carried out systematically by first considering the infinite medium case for monoenergetic neutrons and then extending it to the finite media, multigroup and anisotropic scattering cases. The results are particularly related with the problems and prospects of the neutron-noise studies by excore detectors in fast reactors and would be particularly useful in developing the technology of malfunction detection by neutron-noise methods.

1. INTRODUCTION

Neutron-noise studies possess great potential for physical parameter estimation and malfunction detection in nuclear reactor systems. In Liquid Metal Fast Breeder Reactors (LMFBRs), excore detectors are normally used for monitoring the neutron flux. Therefore, the intervening non-multiplying media such as reflectors, neutronic and thermal shields between the reactor core and the excore detectors are likely to have an effect on the neutron-noise signals reaching the excore detectors. Thus for a correct interpretation of such noise signals, it is essential to understand the physics of the noise transmission characteristics of the above non-multiplying media. This paper provides the methodology and results for such a study. The problem has been studied previously by Singh et al. (1982) and John et al. (1982) using diffusion theory. However, the results of diffusion theory are not reliable for high frequencies and certain noise sources in LMFBRs are of high frequency character. First the study is conducted for an infinite slab case using one energy group for neutrons and then it is extended to a finite medium case and then the multigroup case with anisotropic scattering. It is worthwhile to mention that Akcasu (1967), Brehm (1967), Travelli (1967) and Cassel and Williams (1969) have also studied the results of transport and diffusion theories in connection with neutron wave propagation problems. However we have addressed the problem in a different context, i.e. we have examined whether the wide band noise arising in the core (from boiling of the coolant or vibration of core internals etc.) is transmitted to excore detectors as such without any distortions (with equal attenuations in all frequencies) or it suffers a frequency dependent attenuation by the non-multiplying media intervening the core and the excore detectors. This will help in interpreting better, the excore detector neutron-noise spectra and in identifying the nature of noise source in LMFBR cores.
2.1 Infinite slab and one group analysis

For a slab system with X-axis extending from \(-\infty\) to \(+\infty\), the time dependent, one dimensional, monoenergetic, isotropic scattering neutron transport equation can be expressed as,

\[
\frac{1}{v} \frac{\partial \phi}{\partial t} + \mu \frac{\partial \phi}{\partial x} + \sigma_t \phi(x,\mu,t) = \frac{\sigma_s}{2} \int_{-1}^{1} \phi(x,\mu',t) d\mu' + Q(x,\mu,t)
\]  

(1)

where the symbols have their usual meaning (Bell and Glasstone, 1970; Williams, 1966). Now we assume a sinusoidally varying neutron-noise source to be present at x=0. The actual time dependence of the neutron source will depend on the nature of noise sources existing in the reactor core. However, the above assumption is justified because a wide band neutron-noise source can be expressed as the superposition of several sinusoidal sources of different frequencies and magnitudes. For this the time variation of the neutron flux, \(\phi\), would also be sinusoidal. Therefore, if \(\delta\phi(x,\mu,t)\) is the fluctuating part of the flux, \(\phi = \phi_o + \delta\phi\), then,

\[
\delta\phi(x,\mu,t) = \psi(x,\mu,\omega) e^{i\omega t}
\]  

(2)

The neutron-noise source, \(\delta S(x,\mu,t)\) at x=0 can be written as,

\[
\delta S(x,\mu,t) = \frac{S_o(\omega)}{4\pi} \delta(x) e^{i\omega t}
\]  

(3)

where \(S_o(\omega)\) will be dependent on the reactor core parameters like flux, cross-sections, etc. Expressing eq.(1) in terms of \(\delta\phi\) and then using eqns.(2) and (3) one can write (Jena and Singh, 1985; Jena, 1986),

\[
\mu \frac{\partial \psi}{\partial x} + (\sigma_t + i\omega) \psi(x,\mu,\omega) = \frac{\sigma_s}{2} \int_{-1}^{1} \psi(x,\mu',\omega) d\mu' + \frac{S_o(\omega)}{4\pi} \delta(x)
\]  

(4)

Taking Fourier transform of the above eq. and solving, one will get (Jena and Singh, 1985),

\[
\psi(K,\mu,\omega) = \frac{S_o(\omega)}{4\pi(\sigma_t + i\omega + iK\mu)} \left[1 - \frac{\sigma_s}{2iK} \ln \left(\frac{\sigma_t + i\omega - iK\mu}{\sigma_t + i\omega + iK\mu}\right)\right]^{-1}
\]  

(5)

where

\[
\psi(K,\mu,\omega) = \int_{-\infty}^{\infty} \psi(x,\mu,\omega) e^{ikx} dx
\]  

(6)

The expression for \(\psi(x,\mu,\omega)\), and hence the total flux,

\[
\psi(x,\omega) = 2\pi \int_{-1}^{1} \psi(x,\mu,\omega) d\mu
\]

(7)

can be obtained by taking inverse Fourier transform of eq.(5). It would involve contour integration and the solution would possess discrete part coming from the pole contribution at \(K=K_0\) and the continuum part from the branch cuts arising due to the log function (See eq.5). The final expression for the total flux can be written as (Jena and Singh, 1985),

\[
\psi(x,\omega) = P(K_0,\omega) S_o(\omega) e^{ik_0x} + S_o(\omega) \int_{-\infty}^{\infty} \gamma Q(\tau,\omega) e^{i\tau x} d\tau
\]  

(7)

where

\[
P(K_0,\omega) = \frac{iK_0 \left[(\sigma_t + i\omega)^2 + K_0^2\right]}{4\pi \sigma_s \left[(\sigma_t + i\omega)^2 - \sigma_s (\sigma_t + i\omega) + K_0^2\right]},
\]  

(8)
\[ Q(\gamma, \omega) = \frac{2L^2}{\left[ 2irL - \sigma_0 \left( \ln \left( \frac{a + i\omega / \mu + \gamma} {a + i\omega / \mu - \gamma} \right) \right]^2 + \pi^2 \sigma_0^2 \right] + \pi^2 \sigma_0^2} \]

and \( r \) varies from \( R = \sqrt{\sigma_0 \omega / \mu} \) to \( \infty \), \( L = -\cos \alpha + i \sin \alpha \), \( \alpha = \tan^{-1} \left( \frac{\omega_0}{\omega} \right) \).

The Auto Power Spectral Density (APSD) of the total flux can be calculated using the relation (Bendat and Piersol, 1971),

\[ \text{APSD}(x, \omega) = \psi(x, \omega) \psi^*(x, \omega) \]

It is difficult to obtain the analytical expressions like eqs. (7) to (9) for angular flux, \( \psi(x, \mu, \omega) \). Therefore, a pure numerical approach, similar to that of \( S_N \) method given by Lathrop (1972) was followed. The essential basis of this approach is that the angular distribution of the neutron flux is evaluated along a number of discrete directions. Following this method, the expression for discretised angular flux can be written as (Jena and Singh, 1986),

\[ \psi_{i + \gamma_2, m + \gamma_2} = \frac{1}{|\mu_m| A} \psi_{x, m + \gamma_2} + \alpha' \psi_{i + \gamma_2, m - \gamma_2} + \chi \]

where

\[ \chi = \sum_{m=1}^{M} \sigma_{i + \gamma_2 \psi_{i + \gamma_2, m, \omega}, \psi_{i + \gamma_2, \omega, S_0(\omega), \delta}} \]

\[ \sigma_{i + \gamma_2 \psi_{i + \gamma_2, m, \omega}, \psi_{i + \gamma_2, \omega, S_0(\omega), \delta}} = \sigma_{i + \gamma_2 \psi_{i + \gamma_2, m, \omega}, \psi_{i + \gamma_2, \omega, S_0(\omega), \delta}} \]

\[ \chi = i + 1 \text{ for } \mu_m < 0 \text{ and } i = 1 \text{ for } \mu_m > 0 \],

\[ \alpha' = (\alpha'_{m + \gamma_2, i + \gamma_2} + \alpha'_{m - \gamma_2, i + \gamma_2}) / \omega_m \]

\[ A = A_{i + 1} + A_i \], \( \delta = 1 \text{ for } i = 1 \), and \( \delta = 0 \text{ for } i \neq 1 \).

\( A_i \) is the area of the \( i \text{th} \) spatial mesh, and \( \alpha'_{m + \gamma_2, i + \gamma_2} \) are the curvature coefficients, \( V_{i + \gamma_2} \) is the volume of the \( i \text{th} \) mesh, \( \omega_m \) is the weight corresponding to the direction, \( \mu_m \). Eq. (11) is solved using power iteration technique. In this method the guess flux is assumed and the above term \( \chi \) is computed using these \( \psi \)'s and then new \( \psi \)'s are calculated using eq. (11). For next iteration again \( \chi \) is computed using new \( \psi \)'s and the iteration process is continued till the previous and present fluxes agree within a predetermined value. Then the APSD of the angular flux is calculated using eq. (10).

2.2. Finite Slab and Multigroup Analysis

The general formulation of the problem for finite slab case is same as discussed above; the only difference is that the frequency dependent flux is evaluated by suitable boundary conditions. We have used eq. (11) for evaluating space and angle dependent flux and apply the following boundary conditions for a finite medium,

\[ \psi_{i+1, m} = 0 \text{ for } \mu_m < 0 \text{ (Vacuum)} \]

(12)

and

\[ \psi_{i, 1-m} = \psi_{i, -m} \text{ for } \mu_m > 0 \text{ (Reflective)} \]

(13)

Eq. (12) states that at the outer boundary of finite media, the flux in all the incoming directions is zero and eq. (13) represents the usual neutron...
reflection condition. With eqs. (12) and (13), the eq. (11) is solved by power iteration technique as mentioned in Section 2.1. For multigroup analysis, the neutron transport equation used is,

\[ \frac{1}{\nu_g} \frac{\partial \psi_{g}}{\partial t} + \mu \frac{\partial \psi_{g}}{\partial x} + \sum_{l=1}^{g} \sigma_{l,g} \psi_{g} \left( x, \mu, \mu', t \right) = \frac{1}{2} \sum_{g=1}^{g} \int \sigma_{g 	o g} \left( x, \mu' \rightarrow \mu' \right) \psi_{g'} \left( x, \mu', t \right) \partial \mu' S_{g} \]

where the neutron energy range is divided into groups in such a way that variation of important cross sections within the group is kept reasonably small. The group flux and cross sections are defined by standard procedure (Bell and Glasstone, 1970; Jena, 1986). Following the procedure of Section 2.1 above, eq. (14) is transformed to frequency domain for isotropic scattering case and then discretising the space and angle variables, finite difference form for the flux can be written as (Jena, 1986),

\[ \psi_{g',i+\gamma_{l}, m+\gamma_{l}} \frac{1}{\nu_{g}} \frac{\partial \psi_{g'}}{\partial t} + \mu \frac{\partial \psi_{g'}}{\partial x} + \sum_{g=1}^{g} \sigma_{l,g} \psi_{g'} \left( x, \mu', \mu, t \right) = \frac{1}{2} \sum_{g=1}^{g} \int \sigma_{g 	o g} \left( x, \mu' \rightarrow \mu \right) \psi_{g} \left( x, \mu, t \right) \partial \mu' S_{g} \]

where

\[ \psi_{g'} = \sum_{g=1}^{g} \sigma_{g 	o g} \psi_{g} \]

It can be seen that for each energy group there is a one speed problem with additional down scatter terms from higher energy groups and hence eq. (15) can be solved as eq. (11).

2.3. Anisotropy and Back Scattering Analysis

To incorporate the anisotropic scattering in the formulation, the scattering kernel of transport equation, \( \sigma_{s} (x, \mu' \rightarrow \mu) \) is expressed in terms of Legendre polynomials, i.e.

\[ \sigma_{s} (x, \mu' \rightarrow \mu) = \sum_{l=0}^{\infty} \frac{2l+1}{4\pi} \sigma_{l,s} (x) P_{l} (\mu) \]

where \( \mu' = \mu' \mu \) and \( P_{l} (\mu) \) are the Legendre polynomials. For \( P_{1} \)-anisotropy and following the above procedure, the angular flux in finite difference from can be written as (Jena, 1986),

\[ \psi_{i+\gamma_{l}, m} = \frac{2}{1+\Delta x_{i}} \sum_{m=1}^{m} \psi_{i+\gamma_{l}, m} W_{m} \]

where

\[ \chi = \sigma_{s} (x) \Delta x_{i} = \sum_{m=1}^{m} \psi_{i+\gamma_{l}, m} W_{m} \]

All other symbols have the usual meaning as explained above. If we assume that \( \beta_{1} \) and \( \beta_{2} \) are the fractions for back and isotropic part of scattering kernel, then eq. (16) remains same but \( \chi \) is modified as,

\[ \chi = \beta_{1} \sigma_{s} (x) \psi_{i+\gamma_{l}, m} \Delta x_{i} + \beta_{2} \sigma_{s} (x) \Delta x_{i} \]

3. CALCULATIONS AND RESULTS

Stainless steel, borated conerate and graphite (graphite is chosen since it may be put around the detector in LMFBRs to increase its efficiency), the typical materials intervening the reactor core and the excore detectors in LMFBRs, are chosen for the study. The monoenergetic and 4-group cross-sections data for these materials (used in the calculations) are given in Tables 1 and 2 respectively.
Table 1 - The values of absorption $\sigma_{ao}$, scattering $\sigma_{so}$ and total cross-section $\sigma_{to}$ for graphite, borated concrete and stainless steel

<table>
<thead>
<tr>
<th>Material</th>
<th>$\sigma_{ao}$ (cm$^{-1}$)</th>
<th>$\sigma_{so}$ (cm$^{-1}$)</th>
<th>$\sigma_{to}$ (cm$^{-1}$)</th>
</tr>
</thead>
<tbody>
<tr>
<td>Graphite</td>
<td>3.2E-4 $^*$</td>
<td>0.385</td>
<td>0.385</td>
</tr>
<tr>
<td>Stainless Steel</td>
<td>0.152</td>
<td>0.562</td>
<td>0.714</td>
</tr>
<tr>
<td>Borated Concrete</td>
<td>2.127</td>
<td>0.477</td>
<td>2.604</td>
</tr>
</tbody>
</table>

$^*$ E-4 = $10^{-4}$

Table 2 - Four Group Data Set

<table>
<thead>
<tr>
<th>Group</th>
<th>Energy boundaries (eV)</th>
<th>Velocities (cms/sec)</th>
</tr>
</thead>
<tbody>
<tr>
<td></td>
<td>Upper</td>
<td>Lower</td>
</tr>
<tr>
<td>1</td>
<td>1.45E7</td>
<td>6.75E4</td>
</tr>
<tr>
<td>2</td>
<td>6.75E4</td>
<td>2.04E3</td>
</tr>
<tr>
<td>3</td>
<td>2.04E3</td>
<td>4.10E-1</td>
</tr>
<tr>
<td>4</td>
<td>4.10E-1</td>
<td>Thermal</td>
</tr>
</tbody>
</table>

2. Macroscopic cross-sections (cm$^{-1}$)

<table>
<thead>
<tr>
<th>Material</th>
<th>Group</th>
<th>$\sigma_{a}$</th>
<th>$\sigma_{b}$</th>
<th>$\sigma_{g\to g}$</th>
<th>$\sigma_{g\to g-1,g}$</th>
<th>$\sigma_{g\to g-2,g}$</th>
<th>$\sigma_{g\to g-3,g}$</th>
</tr>
</thead>
<tbody>
<tr>
<td>Graphite</td>
<td>1</td>
<td>2.3E-6</td>
<td>0.260</td>
<td>0.259</td>
<td>1.8E-4</td>
<td>9.5E-3</td>
<td>0.0</td>
</tr>
<tr>
<td></td>
<td>2</td>
<td>0.0</td>
<td>0.377</td>
<td>0.377</td>
<td>4.6E-5</td>
<td>0.0</td>
<td>0.0</td>
</tr>
<tr>
<td></td>
<td>3</td>
<td>1.1E-5</td>
<td>0.377</td>
<td>0.368</td>
<td>4.6E-5</td>
<td>9.5E-3</td>
<td>0.0</td>
</tr>
<tr>
<td></td>
<td>4</td>
<td>3.2E-4</td>
<td>0.372</td>
<td>0.372</td>
<td>9.5E-3</td>
<td>9.5E-3</td>
<td>0.0</td>
</tr>
<tr>
<td>Borated</td>
<td>1</td>
<td>1.2E-3</td>
<td>1.4E-3</td>
<td>1.6E-4</td>
<td>9.7E-3</td>
<td>2.1E-4</td>
<td>4.0E-5</td>
</tr>
<tr>
<td>Concrete</td>
<td>2</td>
<td>9.4E-3</td>
<td>9.7E-3</td>
<td>2.1E-4</td>
<td>4.0E-5</td>
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<tr>
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<td>2.604</td>
<td>0.477</td>
<td>4.1E-3</td>
<td>1.8E-5</td>
<td>0.0</td>
</tr>
</tbody>
</table>

3.1. Infinite Slab and One Energy Group Results

From eq.(7), it can be seen that the frequency dependence of the total flux comes only from the frequency dependence of neutron-noise source and the eigenvalues $K_{w}$. Frequency dependence of the source is not important because flux is just proportional to $S_{0}(w)$. However, $K$ affects the flux through the exponential term, providing exponential attenuation and through the $P(K_{w},w)$ and $Q(v,w)$. If we define an effective attenuation factor, $K_{eff}(x,w) = -dy/dx/V(x,w)$ and normalised $K_{eff}(w) = K_{eff}(x,w)/K_{eff}(x,0)$ and evaluate it for different frequencies and different materials (graphite, borated concrete and stainless steel), we find that $K_{eff}$ is constant for all frequencies, $w < vG_{a}$ and increase with frequency thereafter. This is also observed by John et al.(1982) while treating the problem with diffusion theory. Thus the transport theory considerations do not affect the diffusion theory limit. We evaluated the APSD of the flux for the three materials mentioned above at several mean free path distances from the source and by taking $S_{0}(w) = 1$. For graphite, the results are given in Fig.1. The distances selected are 2, 20 and 40 mean free paths (mfp) from the source and the break-frequencies of the APSD at these distances are found to be 17, 13 and 11 Hz respectively. That is, the break-frequency decreases with distance from the source. Thus the thickness of the medium through which the noise signal propagates may play a significant role in identifying the nature of the noise sources. The peak frequency of the source spectrum representing some physical process may get shifted to lower frequency side. This is depicted very well in Fig.2, where the source function $S_{0}(w)$,
Fig. 1. Variation of APSD with Frequency for Graphite

Fig. 2. APSD of Neutron Flux with Gaussian Noise Source

arbitrarily taken as Gaussian with mean 13 Hz and standard deviation 10, is plotted along with the APSD of the flux in graphite with $S_0(w)$ to be unity and Gaussian. It may be seen that the peak of the source at 13 Hz is revealed at 9.5 Hz in the APSD calculated at 20 mfp distance.

APSDs are also calculated for stainless steel and borated concrete at 2 and 20 mfp distances from the source. Here also, the break-frequency of APSD decreases with distance. The comparison of the calculations with diffusion results indicate then while for graphite, diffusion and transport theory results are similar for stainless steel and borated concrete diffusion theory under predicts the break-frequency.

The APSD of the angular flux at 2 mfp were also calculated for the three materials using 20-point quadrature set in eq.11 and for $\mu = .12060, 0.66302$ and $0.98535$. A typical set of results is given in Fig.3 for borated concrete. It is found that for all the three materials the break-frequency of the APSD of the angular flux depends upon $\mu$; the smaller the angle, sharper is the fall in APSD. Physically the dependence of break-frequency of the angular flux can be explained, based on the distance the neutrons with different $\mu$ travel in the medium. Such a distance being $x/\mu$,
Fig. 3. Variation of APSD of Angular and Total Flux with Frequency for Borated Concrete

indicates that the neutrons with smaller $\mu$ travel larger distance inside the medium and hence as observed in the case of space dependent total neutron flux, smaller is the corresponding break-frequency. Thus, the preferential monitoring of the forward peaked neutron flux may provide minimum distortion in the neutron-noise source spectrum and hence a better interpretation of the neutron-noise experiments by excore detectors in LMFBRs. However, smaller the $\mu$, smaller will be the neutron-noise flux and hence difficult to detect.

3.2. Finite Slab and Multigroup Results

Numerical calculations were performed for finite sizes of graphite and borated concrete using a 20-point quadrature set. The frequency dependent behaviour of APSD of neutron-noise flux for different locations for graphite is shown in Fig. 4. The broken and solid lines represent the infinite and

Fig. 4. Variation of APSD with Frequency for Graphite

finite media cases respectively. It can be seen that finite media considerations lead to increase in break-frequencies of the APSD of neutron noise flux. The physical explanation for this can be given in terms of neutron leakage that exists as an additional loss of neutrons in finite media and the fact that neutrons undergo less collisions before detection resulting in less distortion of frequency spectrum. Also in a finite medium, the effective lifetime of the neutrons decreases. Therefore, the break-frequency increases as long as spacing of successive neutron pulses is
greater than the maximum neutron lifetime in the medium (For details see John et al. 1982 and Jena 1986). Practical implication of the above result is that a suitable design adjustment in the size of the neutron-noise transmitting medium, can lead to better and simpler interpretations of the neutron-noise experiments.

The multigroup calculations were performed with four groups. In multigroup case it is more meaningful to characterise the fluctuations in detector response rather than the neutron flux. Therefore, the results were obtained for BF$_3$ and U$_{235}$ detector responses and are presented in Fig.5 for graphite. The APSD of the detector responses have been calculated using the formula

\[ \text{APSD}_{d} = \frac{1}{N} \sum_{j} \sigma_{ag}^{d} \psi(L,\omega)^{2} \]

where \( N \) is the normalisation constant, \( \sigma_{ag}^{d} \) is the detector absorption cross-section and \( L \) is the detector location. It can be seen from the Fig.5 that the detector responses are close to the thermal group since these two detectors have higher response to lower energy neutrons. In terms of flux, the break-frequency of the APSD of energy dependent neutron flux, is more for high energy neutron flux and the break-frequency of the APSD of total flux is closer to that of high energy neutron flux.

3.3. Anisotropy and Back Scattering Results

Detailed calculations were performed for P$_1$ anisotropic scattering using 20-point quadrature set for 2 mfp thick graphite and borated concrete. The anisotropic cross-section data were derived from DLC2 set (Wright, 1972). The calculations revealed that both in graphite and borated concrete, the break-frequency of APSD of neutron flux decreases with P$_1$ anisotropy. To understand this physically, we did S$_1$-analysis (Jena, 1986) which indicated that the effective absorption of neutrons is less in forward scattering and so the effective lifetime of the neutrons increases. This will decrease the break-frequency of APSD of neutron flux with anisotropic scattering. For back scattering case also the calculations were made using again 20-point quadrature set. For similar case analytic results had been derived by Singh (1971). However, the analytical expressions are complicated and do not reveal the trend straightforward. Thus the numerical calculations were performed with different \( \beta_1 \) and \( \beta_2 \). As expected from forward scattering analysis, here break-frequency increases with increase in the fraction of back-scattering. Back scattering analysis is useful for crystalline media like graphite in which the neutrons do undergo complete back scattering corresponding to the energy given by Bragg condition.
4. CONCLUSIONS

One can draw the following conclusions from this study.

1. The neutron-noise transmission characteristics predicted by transport theory are qualitatively similar to the diffusion theory predictions. The $\mu_0 c_0 / (\Gamma \gamma_0)$ limit of the attenuation parameter being independent of frequency by diffusion theory predictions remains unaltered even under transport theory treatment.

2. The break-frequency of the APSD of neutron flux decreases with distance from the source. This indicates that longer the distance travelled by the neutron-noise signals, more severe is the neutron-noise spectrum distortion.

3. For graphite, transport and diffusion theories predict the same break frequency whereas for stainless steel and borated concrete diffusion theory underpredicts it as compared to the transport theory.

4. In systems where graphite is used to increase the efficiency of the detectors by thermalising the neutrons, the peak frequency of the predominant noise source representing the physical processes like boiling or mechanical vibrations of the core internals in LMFBRs will get shifted to lower frequency side; the shift being dependent on the thickness of the graphite used around the detector. But this is not so for other materials like stainless steel or borated concrete because the break-frequency for these materials is quite high.

5. The break-frequency of the APSD of the angular neutron flux is $\mu$-dependent. Smaller the angle, sharper is the fall in APSD of the angular flux and hence showing smaller break-frequency. Thus preferential monitoring of the forward peaked neutron flux fluctuations would provide minimum distortion in the neutron-noise source spectrum.

6. Finite media considerations lead to the increase in the break-frequency of the APSD of the neutron flux fluctuations. Thus a proper optimisation of the size of the moderating materials around the detectors is needed to find suitable balance between the extent of neutron thermalisation and the break-frequency needed.

7. The multigroup considerations in the study of the break-frequency of the APSD of the neutron flux indicate that high energy neutron flux shows larger break-frequency. But the break-frequencies of the detector responses are close to that of low energy neutron flux.

8. Anisotropic scattering considerations lead to decrease in the break frequency of the APSD of the neutron flux. However, the decrease is small for materials like graphite and borated concrete. Isotropic neutron scattering approximation is a good approximation for studying neutron-noise transmission characteristics through non-multiplying media in LMFBRs.

REFERENCES


SPATIAL SENSITIVITY OF A 900 MWe PWR IN-CORE DETECTOR SIGNAL TO FUEL DISPLACEMENT AND VOID FRACTION

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ABSTRACT - In core detector signal fluctuations have proved to be efficient for the detection and the characterisation of some phenomena such as vibrations of fuel assemblies, control rods or thimbles, and more rarely, ebullition.

Hereafter is studied the quantification of the eventual anomaly found from the measurements of such fluctuations. The question is : assuming a hypothesis about the nature of the anomaly is it possible to precise its level (i.e. in term of vibration amplitude or boiling rate etc...).

In case of vibration of a fuel assembly containing a detector, the quantification problem is to be solved easily. This won't be the case for other fuel assemblies vibrations or for the ebullition phenomena. A simple solution generally won't be found.

The detector sensitivity spatial repartition in regard of such perturbations is studied and it is shown that despite the very sharp shape of that repartition (with a maximum at the detector location) the seeing length of the detector can be large for the phenomena to be currently considered in a PWR.

INTRODUCTION

The quantification problems linked to the measurements of in-core detectors signal fluctuations are seldom studied on PWRs. In-core signal fluctuations have for the most part shown to be efficient for the surveillance of some core components vibratory behavior (fuel assemblies, /l/ to /4/, control rods /5/, and more recently, thimbles /6/, /7/).

Hence this vibratory surveillance concerns materials which are to be (or may be) regularly changed. Furthermore, the quantification of the abnormal phenomena appears not to be an information that the operator urgently needs since for him the warning is given by the primary circuit activity level. However this information can be helpful. when a choice, between several hypotheses concerning the nature of the detected event, is to be made. In fact, an in-core detectors are distributed in the core, an anomaly characterisation needs some hypothesis about the propagation of the flux perturbation from its assumed origin to the detector locations. The full characterisation of an anomaly (or of a new event on NPSD(*) signatures) from in-core signal fluctuations generally needs the following stages to be solved :

- qualitatively characterise the event by the examination of the main frequencies, the NPSD evolution versus the detector locations, the phase and coherence laws between signals from several detectors,
- add some technological informations (about primary circuit, refueling...),
- a first hypothesis for the nature of the event is then obtained,
- consider the quantification problem and compare the rms values which were to be obtained according to the previous hypothesis, to the ones measured,
- if there is agreement, the hypothesis is enhanced, otherwise an other one is to be found.

* Signals are supposed to be transformed in normalised power spectral density.
Then, the solution to the quantification problem won't often lead to a precise value for a vibratory amplitude, or a boiling rate etc... but will provide an answer to questions such as the following one: under the hypothesis that an "x" event occurs in point \( P(x,y,z) \) with an intensity \( A(t) \), will the effect on the signal rms value of a detector located in \( (x_0,y_0,z_0) \) be compatible with the one observed?

When the phenomena to be considered is fuel assemblies vibrations or boiling the spatial Green functions which provide an answer to such questions are calculated hereafter.

1. FUEL ASSEMBLIES VIBRATION

A distinction has to be made between the vibration of an assembly which contains a detector (case n° 1) and the one of an assembly which doesn't (case n° 2).

Case n° 1

With the hypothesis that the detector doesn't move with the fuel assembly the expression of \( \frac{\Delta E}{s} = f(6x) \) (where \( s \) is the detector signal and \( 6x \) the vibration amplitude) has been calculated /8/. It consists in two terms: one (first order) is due to the \( GM \) macroscopic gradient created by the fuel pins surrounding the detector tube guide; the other (second order) is due to the fine flux distribution taking into account the materials surrounding the fission chamber. It has been obtained:

\[
\frac{\Delta s}{s} (6x) = a \cdot 6x^2 + b \cdot 6x,
\]

\( a = -0.12 \)

\( b = GM \)

The basic hypothesis according to which the detector usually doesn't move in its thimble has been tested on a device equipped with a fuel assembly of the 1300 MW PWR series type. Hydraulic conditions of the mock-up well represented the reactor ones. As shown in figure /1/, a very small accelerometer was attached inside the thimble, while a laser vibrometer pointed the grids, giving then the information about the fuel assembly movement. The coherence function between the 2 signals takes a high value (\( \geq 0.7 \)) when the frequency is around 18 Hz. It shows that for frequencies lower than 18 Hz the thimble (and the detector by extension) doesn't follow the fuel assembly movement. Results obtained from this mock-up are supposed to represent also the detector dragging in a fuel assembly of a 900 MW PWR type whose eigen frequencies are not very far from the one of a 1300 MW PWR fuel assembly /10/.

Case n° 2

This part deals with the seeing length of an in-core detector in regard of the vibration of fuel assemblies in the neighbourhood.

It will first be defined a spatial weighting function, \( W(r) \), which is the signal normalized variation \( \frac{\Delta E_s}{E_0} \) due to a unity perturbation occuring at the point \( r \).

The sensitivity map \( S(r) \) will then be deduced from \( W(r) \): let \( p \) be a perturbation, \( S(r) \) is the signal normalized variation due to the \( p \) perturbation located at the point \( r \).

Below are drawn the calculation of \( W(r) \) and \( S(r) \) for \( p \) being fuel assemblies vibrations. These are based upon the first order transport equation perturbation and are similar to those of ref. /9/ developed for the boiling in BWR.

First, the transport operator is expressed in the frequency domain \( (\mathcal{H}) \), with the delayed neutrons spectra taken equal to the neutrons spectra in static conditions. Then the well known fact according to which the frequencies to be considered lie in the plateau region of the reactor transfer function is used. \( \mathcal{H} \) can then be written in the following way:

\[
(1) \quad \mathcal{H} = \hat{\mathcal{N}} V \cdots + \bar{P}_a \cdots - \int_0^\omega \int_{\mathcal{A}_H} \bar{P}_s (E' \rightarrow E ; \Omega' \rightarrow \Omega) \cdots dE' d\Omega' \\
- \frac{1}{4\pi} x_{at} (E) (1-\beta) \int_0^\omega \int_{\mathcal{A}_H} \nu \bar{P}_{E'} \cdots dE' d\Omega'
\]
In fact for the considered frequencies, this expression is \( \omega \) independent, and to simplify the writing, the subscript \( \omega \) will now be omitted. It is obtained:

\[
(2) \quad \delta H = \delta \bar{E} a \ldots - \int_0^\infty \int_{4\pi} \delta \bar{E} \ldots \ dE' \ d\Omega' - \frac{1}{4\pi} \chi_{st} (E) (1-\beta) \int_0^\infty \int_{4\pi} \delta v \bar{E} \ldots \ dE' \ d\Omega'
\]

The fluctuation of the signal is:

\[
\delta s = \langle \xi_d, \delta \psi \rangle = \langle S, \psi^+ \rangle
\]

where \( \langle \ldots \rangle \) stands for \( \int_{R} \int_0^\infty \int_{4\pi} \ldots \ d\xi \ dE \ d\Omega \)

\[
\delta s = - \delta H v_o
\]

\( \psi^+ \) is solution of \( H^+ \psi^+ = \xi_d \)

with \( \xi_d \) = fission cross-section of the detector

and \( H^+ \) adjoint operator of \( H \) written as below:

\[
(3) \quad H^+ = - \bar{N} \ldots + \bar{E} a \ldots - \int_0^\infty \int_{4\pi} \bar{E} s (E\rightarrow E'; \Omega\rightarrow\Omega') \ldots \ dE' \ d\Omega'
\]

\[
- \frac{1}{4\pi} v \Sigma_f (E) (1-\beta) \int_0^\infty \int_{4\pi} \chi(E') \ldots \ dE' \ d\Omega'
\]

And \( \delta s \) is obtained according to \( \delta s = - \langle \delta H v_o, \psi^+ \rangle \). \( \delta H \) is now to be precised for a specified \( p \) perturbation that is chosen here as a fuel assembly vibration amplitude.

\[
(4) \quad \delta H = - \frac{dH}{dp} \delta p \quad \text{and} \quad W_p (r) \quad \text{is then defined as:}
\]

\[
(5) \quad W_p (r) = - \int_0^\infty \int_{4\pi} \psi^+_o (r) \frac{dH}{dp} \psi_o (r) \ dE \ d\Omega
\]

which is the weight function expressing the variation \( \delta s(r) \) due to \( \delta p (r) \) set equal to \( 1 \) on a unit volume including \( r \), or Green function of the perturbation \( p \):

\[
(6) \quad \delta s = \int_{R} \delta p (r) \cdot W_p (r) \ d\Omega
\]

(*) Considering a calculation domain \( R \) on which surface, \( \psi^+ \) is weak enough so that

\[
\int_S \int_{4\pi} \psi^+ \delta \Omega_\ldots \ d\Omega \ d\Omega
\]

which is to appear in \( \delta s \) can be neglected.
In meshes $r_1$ and $r_2$, the materials proportion is the following:

<table>
<thead>
<tr>
<th></th>
<th>Mesh $r_1$</th>
<th>Mesh $r_2$</th>
</tr>
</thead>
<tbody>
<tr>
<td>Mat.1</td>
<td>1</td>
<td>0.5</td>
</tr>
<tr>
<td>Mat.2</td>
<td>0</td>
<td>0.5</td>
</tr>
<tr>
<td>Stage 0</td>
<td>0</td>
<td>1</td>
</tr>
<tr>
<td>Stage 1</td>
<td>1</td>
<td>0.75</td>
</tr>
<tr>
<td></td>
<td></td>
<td>0.25</td>
</tr>
</tbody>
</table>

From the above table, linearised coefficients set as $a_1$ and $a_2$, which are the materials proportion variation for unit displacement $\delta x$ can be calculated. They are the following:

<table>
<thead>
<tr>
<th></th>
<th>Mesh $r_1$</th>
<th>Mesh $r_2$</th>
</tr>
</thead>
<tbody>
<tr>
<td>$a_1$</td>
<td>$\frac{1}{\delta x}$</td>
<td>$\frac{1}{\delta x}$</td>
</tr>
<tr>
<td>$a_2$</td>
<td>$\frac{1}{\delta x}$</td>
<td>$\frac{0.25}{\delta x}$</td>
</tr>
</tbody>
</table>

In fact these linearised coefficients are only to be used for displacement values close to the $\delta x$ value with which they have been obtained. Hence they are now written as $a_1, \delta x$ and $a_2, \delta x$.

Let $H_1$ now be the $H$ operator with cross-sections from material n° 1, and $H_2$, be $H$ with cross-sections from material n° 2.

$$\frac{dH_1(\tau)}{dx} = a_1, \delta x \quad H_1(\tau)$$ and $$\frac{dH_2(\tau)}{dx} = a_2, \delta x \quad H_2(\tau)$$

It is expressed so, that the 3 cross-sections of $H$ (absorption, scattering and fission), are changed all the same way in the mesh $\tau$.

Weight functions can then be defined so:

(7) $$W_1(\tau) = -a_1, \delta x \int_0^\omega \int_4 \psi^+ H_1(\tau) \psi_0 d\Omega dE$$

From which we get:

(8) $$\delta s_1 = \int_\tau \delta x (\tau) \quad W_1(\tau) \quad d\tau$$

For $i = 1, 2$.

Figures /2/ and /3/ show some spatial cuts of the functions $W_1(\tau)$ and $W_2(\tau)$ with $a_1, \delta x$ and $a_2, \delta x = 1$. Material n° 1 is composed of 2.1 % enriched fuel, while it is 3.1% enriched fuel for material n° 2. Other components are water and clads. These are homogeneised with the fuel on an assembly cell.

$\psi_0^+$ and $\psi^+$ of eq (7) have been obtained from 2D(R-Z) transport calculations which conditions are exposed in the annexe.

The next stage is to build the sensitivity function $S(\tau) = \frac{\delta s(\tau)}{so}$
From it, it is possible to obtain the value \( \frac{\delta s}{so} (I,Z) = S_I (Z) \) which is the relative variation of the detector signal due to a \( \delta x \) displacement of a part (at the level \( Z \)) of the I fuel assembly. It has been calculated as follows:

\[
S_I (Z) = \frac{1}{so} \int \delta x (R_{I-1},Z) - \delta x (R_I, Z) \, dR - \delta x (R_{I-1},Z) - \delta x (R_I, Z)
\]

or

\[
S_I (Z) = \frac{1}{so} \int \delta x (Z) \, dV (I-1,Z) - \delta x (Z) \, dV (I,Z)
\]

with \( \delta x (Z) \) being the moved volume of the I fuel assembly at a level \( Z \).

The result of these \( S(I,Z) \) calculations is reported in figure 4/4 where the parameters of the perturbation are the following: the moved piece of fuel is at the level \( Z \) of the I fuel assembly border, it is \( 1 \) cm high and is displaced by \( \delta x = 0.1 \) mm. The coordinates \( (R(I), Z) \) of this piece are quoted in a repair which centre is the in-core detector as figured below:

The results for \( S(I,Z) \) obtained in the R-Z configuration (see the annex) can be used for a detector located anywhere in the reactor core under the restriction that it has to be far (\( \geq 2 \) assemblies) from the border line. This, because the cross-sections have been obtained by homogenisation on an infinite critical lattice.

It is now possible to define a shape \( \delta x (I,Z) \) for the displacement of the I assembly on its full length and calculate the effect \( \frac{\delta s}{so} (I) \) on the detector signal. This has been carried out for a vibration of amplitude \( \delta x = 0.1 \) mm, which maximum is located at the level of the detector \( (Z = 0) \) (shape of first eigen mode). The following peak values have been found:

<table>
<thead>
<tr>
<th>Vibration of an assembly I</th>
<th>I=2</th>
<th>I=3</th>
<th>I=4</th>
<th>I=5</th>
</tr>
</thead>
<tbody>
<tr>
<td>( \frac{\delta s}{so} )</td>
<td>( 2.8 \times 10^{-3} )</td>
<td>( 2.6 \times 10^{-3} )</td>
<td>( 2.0 \times 10^{-3} )</td>
<td>( 1.1 \times 10^{-3} )</td>
</tr>
</tbody>
</table>

It is to be the rms value in the range \( 0.7 \times 10^{-3} \) to \( 0.3 \times 10^{-3} \) (peak at \( 4\sigma \)) which can be compared to the range obtained in 4/4. This result also shows that the detector seeing length for fuel assemblies vibration is rather large. This last property is a consequence of the \( W \) functions shape: beyond some \( 10 \) cm (the first half assembly) it becomes almost flat and there is slow evolution between the weight values for the 2nd assembly meshes and the ones of the 4th assembly meshes (the values for the 5th assembly include border effects).
2. EBULLITION

In that case the same tools as in the previous example are used. The perturbation $p$ is then a variation of the material cross-sections due to the evolution of the water density.

For homogeneised fuel assembly cross-sections and a water density variation $\frac{\delta p}{\rho_0}$ we have:

$$\frac{\delta a}{\lambda a} = a \frac{\delta p}{\rho_0}, \quad \frac{\delta s}{\lambda s} = b \frac{\delta p}{\rho_0}, \quad \frac{\delta \langle v_E \rangle}{v_E} = c \frac{\delta p}{\rho_0}$$

The coefficient $a$, $b$, $c$ (Table 1) are energy and $\frac{\delta p}{\rho_0}$ dependent. They have been calculated for the 5 energy groups needed in $\Psi$ and $\Psi^+$ calculations, and for $\frac{\delta p}{\rho_0} = -10\%$, $-50\%$, $-86\%$ (that is to say for boiling rate of 1.85%, 16% and 100%).

$a$ and $b$ are little sensitive to $\frac{\delta p}{\rho_0}$ while $c$ is very sensitive. Again, for low energies $b$ is close to 1 since water has an effect mainly on scattering. The operator $\delta H$ to be considered here is (see Eq (2)):

$$(10) \quad \delta H \frac{\delta p}{\rho_0} = \left[ a \frac{\delta p}{\rho_0}, E \right] \Sigma_a (E) \ldots \int_{E_1} b \left( \frac{\delta p}{\rho_0}, E' \right) \int_{4\pi} F_o \left( E' \rightarrow E ; \Omega' \rightarrow \Omega \right) \ldots dE' d\Omega'
- \frac{1}{4\pi} \chi_{st} (E) (1-B) \int_{E_1} c \left( \frac{\delta p}{\rho_0}, E' \right) \nu E_f (E') \int_{4\pi} \ldots dE' d\Omega'$$

According to (5) $W_p (\vec{r})$ was calculated:

$$W_p \left( \vec{r} \right) = W_{\delta p/\rho_0} \left( \vec{r} \right) = -\int_0^\infty \int_{4\pi} \Psi_0 \left( \vec{r} \right) \frac{dH}{dp/\rho_0} \cdot \Psi \left( \vec{r} \right) dE d\Omega$$

and

$$-\frac{\delta s}{s_0} \left( \vec{r} \right) = \frac{1}{s_0} \left[ \frac{\delta p}{\rho_0} \left( \vec{r} \right) W_{\delta p/\rho_0} \left( \vec{r} \right) \right]$$

which gives the normalized detection signal variation for a change $\frac{\delta p}{\rho_0}$ of the water density, in a 1 cm$^3$ volume centred at $\vec{r}$ ($R, Z$).

In fact the part of $\delta s \left( \vec{r} \right)$ due to the fission cross-section evolution is to be neglected in regard of the one due to other cross-section perturbations.

$W \frac{\delta p}{\rho_0} \left( \vec{r} \right)$ is then almost linear versus $\frac{\delta p}{\rho_0}$.

The results show in figure 5/ are cuts of $W_{\delta p/\rho_0} \left( \vec{r} \right)$ for $\frac{\delta p}{\rho_0} = 86\%$ (or $a = 100\%$) in a unit volume containing $\vec{r}$. The point $\vec{r}$ is defined by the $R-Z$ coordinates of the geometry used for transport calculations.

The shape of this weight function (mainly due to the one of the adjoint flux (see fig. 6) is very sharp but it doesn't mean, for PWR, that only the boiling occurring very near of the detector can be detected. In fact, on a PWR the eventual boiling is strongly located and a boiling rate at a far distance of a detector is likely to be detected only if no boiling or other unknown noise is occurring in the detector close neighbourhood.

3. SEEING LENGTH OF AN IN-CORE DETECTOR

The very sharp shape of the weight function cuts, leads to a similar shape for the seeing length of the in-core detectors only if the perturbation distribution is approximately uniform in the reactor core. This is the case of boiling in BWR and it is generally not in PWR. In these reactors the frequent non-coincidence between the perturbation location and the detector close vicinity is enough for the perturbation to be detected roughly the same way by several far located detectors. It is the case for fuel assemblies vibrations and it has been observed in 1/4/ that such perturbation occurring on the external assemblies was easily detected by centre-located detectors (see fig. 7).

It could also be the case for boiling which usually is likely to take place at the fuel assembly border (where are the hot points).
The following conclusion can be drawn for perturbations usually occurring in a PWR lattice, that are to be detected by in-core detectors signal fluctuations and which frequency range lies in the plateau region of the reactor transfer function: the seeing length of the detector is often large, though the spatial distribution of its sensitivity is sharp.

CONCLUSION

The quantification of a perturbation located in the core, from its effects on in-core detector signals is a problem whose solution, usually won't be unique.

Neutron fluctuations study is devoted to weak intensity perturbations and the associated rms values spatial distribution can be set as the product of the intensity (or amplitude) and the spatial Green function which cuts have been drawn for two phenomena.

Practically, the number of eventual in-core detector locations in the core permits, if not an exact localisation of the perturbation, at least a reduction of the spatial areas where it is to be sought.

From a reduced set of likely locations, a reduced set of the perturbation intensity values which were the only ones to be compatible with the measured detector signal rms values could also be found.

ANNEXE

\( \Psi \) and \( \Psi^+ \) calculations.

Forward flux of eq. (5) is critical flux. Adjoint flux is solution of adjoint operator to be associated with a reactor core subcritical by \( \beta \), supplied with an isotropic source having the spatial location of the in-core detector and the energy distribution intensity of an US fission cross-section. These fluxes have been obtained through 2D R-Z transport calculations carried out with the DOT code /11/.

Energy meshing : 5 groups (see limits in table 1).

Angular meshing : 16 directions.

Spatial meshing : the maximum radius is 108.8 cm corresponding then to 4 crowns 21.42 cm wide (an assembly side) surrounding a cylindrical central fuel assembly (\( r = 12.08 \) cm). The remaining \( \pm 10 \) cm wide zone is a transition area before the border. 43 non regular meshes are used.

The height is 83 cm above and below the detector (reflection symmetry at the detection plane) and is described by 30 irregular meshes.

Precisions : As the sensitivity distribution is to be obtained from subsequent subtractions (Eq. 9) a high degree of precision for \( \Psi \) and \( \Psi^+ \) is required. The respected convergence criteria is \( 5.10^{-6} \) (internal and external iterations). Then the following equality : \( \langle Q, \Psi \rangle = \langle Q^+, \Psi \rangle \) or \( \tau^+ = \tau \) was verified by \( \frac{\tau^+ - \tau}{\tau} < 4.10^{-4} \)
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/10/ F.R. MYNATT - F.J. MAC KENTHALER - P.N. STEVENS
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**FIGURE 1**

**W_1 RADIAL CUTS**
- \( R : 1.3 \text{ cm to } 107 \text{ cm} \)
- Normalized values to \( W_1(0,0,0) = 1.07 \times 10^{-1} \)
- \( \phi_0 = 1 \text{ cm} \)
- \( \phi_0 = 0.075 \text{ cm} \)

**W_2 RADIAL CUTS**
- \( R : 1.3 \text{ cm to } 107 \text{ cm} \)
- Normalized values to \( W_2(0,0,0) = 0.725 \times 10^{-1} \)

**FIGURE 2**

**W_1 AXIAL CUTS**
- \( Z : 2.17 \text{ cm to } 81.3 \text{ cm} \)
- Normalized values to \( W_1(0,0,0) = 1.07 \times 10^{-2} \)
- \( \phi_0 = 1 \text{ cm} \)
- \( \phi_0 = 0.075 \text{ cm} \)

**W_2 AXIAL CUTS**
- \( Z : 2.17 \text{ cm to } 81.3 \text{ cm} \)
- Normalized values to \( W_2(0,0,0) = 0.725 \times 10^{-1} \)

**FIGURE 3**

PNE-K
METHOD OF MONITORING REACTOR SYSTEM 
BY STATE SPACE TRAJECTORY PATTERN

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Ukishima-cho, Kawasaki-ku, Kawasaki-city 210 Japan

Abstract — A method of monitoring system states by a state space trajectory pattern is proposed. This method is applied to a one-dimensional system, a two-dimensional system and reactor systems with temperature or void feedback using computer simulation. It is shown that the resultant deterministic and stochastic trajectory patterns correspond closely to each other. This method permits direct treatment of raw data and also nonlinear system and detection of abnormal condition indicated in either two- or three-dimensional space.

1. INTRODUCTION

In reactor noise field, raw noise data in a reactor system are essential and have been analyzed by various highly efficient methods by which spectral density, correlation function, coherence function, auto-regressive coefficient etc. are obtained. Before analyzing the raw data, it is also important to observe carefully how the data behaves as time passes. Amplitude itself and rate of change or time derivative of the signal are the most basic factors in this case. The simplest method of visualizing raw data is time record plotting. Here, as another visual way, a two-dimensional expression is devised like Lissajous figures which were ever used on the analysis of neutron kinetics (Seifritz, 1966) and RPV vibration (Wach, 1977) Plotting amplitude of a signal versus its time derivative in a two-dimensional space will make a particular pattern corresponding to a system state, like a deterministic phase space analysis. From this viewpoint, the following two propositions can be presented (Nomura, 1986)

(1) The n-dimensional state equation for dynamic systems is written in the form

$$\frac{dX(t)}{dt} = f(X(t), U(t), t),$$

where $X(t)$ is the output vector $(x_1, x_2, ..., x_n)$ of the system at time $t$, and $U(t)$ the internal or external input vector $(u_1, u_2, ..., u_m)$ into the system at that time, where $m \leq n$. An n-dimensional state space is one defined by $n$ axes $x_1, x_2, ..., x_n$ at time $t$. If $dx_1/dt, dx_2/dt, ..., dx_n/dt$ axes are added, a projection of this new 2n-dimensional state space becomes the ordinary state space.

(2) In most naturally-occurring dynamic systems, signals with continuous variation are observed. An example is the output noise signal emitted by the system. It can be expected that, if the trajectory of such a signal of continuous variation is depicted in the multi-dimensional state space defined above, as well as in ordinary state space, a trajectory pattern (shape, density, size and movement) corresponding to the system state will be obtained. This pattern and system state are in one-to-one correspondence. Since problems are difficult to concretize in more than 3 dimensions, the pattern will be treated in two- or three-dimensional space, which can be considered to represent the projection of one depicted in higher-order dimensional space.
Two propositions above were applied to one-dimensional system, two-dimensional system and reactor systems with temperature feedback or void feedback as a first step by computer simulation.

2. TRAJECTORY PATTERN IN DISCRETE SYSTEM

In a continuous system, the trajectory pattern depicted in phase plane \((x, dx/dt)\) is definitely determined, if the input \(u(t)\) is definite. However, in a discrete system, the trajectory pattern has to be depicted in the plane \((x, \Delta x/\Delta t)\), which depends on sampling time \(\Delta t\). Particularly, in the noise field, analysis in discrete system plays an important role due to the lack of information on internal noise. So, it is worthwhile to study, as the first step, the trajectory pattern in a discrete system, in case of deterministic system.

For example, the system described by 1st order differential equation

\[
\frac{dx(t)}{dt} = -c \cdot x(t) + u(t)
\]  

(1)

is taken into account. If \(u(t)\) is impulse or the corresponding initial value \(x(0)\), the continuous trajectory pattern is depicted in phase plane \((x, dx/dt)\) like Fig.1(a), where continuous trajectory is observed. In this system, assume that sampling time \(\Delta t\) is 0.5 second. Then, a discrete pattern with slightly different gradient is observed in discrete space \((x, \Delta x/\Delta t)\) like Fig.1(b). In case of 2nd order differential equation

\[
\frac{d^2x(t)}{dt^2} = -k \cdot x(t) + u(t),
\]  

(2)

the trajectory pattern is obtained in phase plane \((x, dx/dt)\) like Fig.2(a), if \(u(t)\) is impulse or the corresponding initial condition \(x(0)\) and \(\dot{x}(0)\). Similarly, a slightly different discrete pattern is observed when sampling time is chosen as 0.02 second. Assuming that \(u(t)\) is Gaussian white noise with standard deviation S.D., then stochastic patterns are observed like Fig.1(c) and Fig.2(c), which respectively correspond well with the discrete deterministic patterns. Thus, when the sampling time is moderately chosen, depending on system parameters, the obtained discrete trajectory pattern is fairly similar to a continuous trajectory pattern. In this sense, the expression \(dx/dt\) or \(\dot{x}\) is adopted, instead of \(\Delta x/\Delta t\), even in case of a stochastic pattern mentioned in later sections.

Fig.1. Continuous and discrete pattern of one-dimensional system. 
(a) Deterministic pattern. (b) Deterministic pattern. (c) Stochastic pattern. 
(Continuous) (Discrete) (Discrete)

Fig.2. Continuous and discrete pattern of harmonic oscillator. 
(a) Deterministic pattern. (b) Deterministic pattern. (c) Stochastic pattern. 
(Continuous) (Discrete) (Discrete)
3. ONE-DIMENSIONAL SYSTEM

A one dimensional linear system is written in the form

\[
\frac{dx(t)}{dt} = -c \cdot x(t) + u(t)
\]  

(3)

where \( x(t) \) is the output from the system at time \( t \) and \( u(t) \) is the external or internal input at that time. Relating to the Proposition(1), if the \( dx(t)/dt \) axis is added, the trajectory pattern depicted in two-dimensional space \( (x, dx/dt) \) is obtained. The results are already shown in Fig.1. As the figure shows, the stochastic pattern corresponds to a deterministic pattern, though the stochastic pattern fluctuates around the average slope.

4. TWO-DIMENSIONAL SYSTEM

The two-dimensional linear system is written in the form

\[
\frac{d^2x(t)}{dt^2} = -k \cdot x(t) - h \cdot \frac{dx(t)}{dt} + u(t)
\]  

(4)

where \( x(t) \), the output from the system at time \( t \), \( k \), spring constant, \( h \), friction coefficient and \( u(t) \) is internal or external input. The input chosen is impulse or the corresponding initial values \( x(0) \) and \( \dot{x}(0) \) in a deterministic case and is Gaussian white noise in a stochastic case, where \( x(0)=0 \) and \( \dot{x}(0)=0 \), these values being adopted in the absence of any inherent limitation to the choice of initial value. Figure 3, relating to the Proposition (1), shows the results obtained in terms of deterministic patterns. Figure 4, relating to the Proposition (2), shows the stochastic patterns corresponding to Fig.3. The patterns of the bottom in both figures show the 2nd order differential system described by Van der Pol equation. The three-dimensional expression is shown on the right side in each figure. The resulting pattern in Fig.4 corresponds very well with that in Fig.3.

![Figure 3](image1.png)

**Fig.3.** Deterministic patterns of two-dimensional systems.

![Figure 4](image2.png)

**Fig.4.** Stochastic patterns of two-dimensional systems.

Here, the fluctuation around the average trajectory of simple limit cycle, obtained in the system described by Van der Pol equation, is studied in more detail. Assume that input noise \( u(t) \) is Gaussian white noise with time mesh \( H \) second and standard deviation S.D., the resultant stochastic patterns are shown in Fig.5. In any case, the average trajectory corresponds with the trajectory of deterministic pattern, while the fluctuation depends on \( H \) and S.D. It is understood that a point corresponding to this system state is willing to go to the same trajectory as that obtained in a deterministic case, that is, limit cycle, but the fluctuation
increases as $H$ and S.D. increase. As another interesting example, the authors study the behaviour of a bi-stable system:

$$\frac{d^2 x(t)}{dt^2} = -b \cdot x(t) \cdot (x^2(t) - p) - s \cdot \frac{dx(t)}{dt} + u(t)$$

(5)

where $b=4$, $p=9$, $s=1$. As Fig.6 shown, in deterministic pattern, the stable point is in either side left or right, depending on initial condition, whereas in stochastic pattern, one eye appears in the case of smaller stochastic input and two eyes appear in the case of larger stochastic input due to the increase of probability of jumping potential barrier.

Fig.5. Stochastic pattern dependence on input noise standard deviation and time mesh in the Van del Pol equation.

Fig.6. Deterministic and Stochastic pattern of bi-stable system.
5 REACTOR SYSTEM

5.1 Reactor system with temperature feedback

The state equation for a reactor system with temperature feedback is expressed in the form

\[
\frac{dN}{dt} = \frac{(\rho - \beta)}{\frac{\lambda}{\ell}} N + \lambda C
\]

(6)

\[
\frac{dC}{dt} = \frac{\beta}{\frac{\lambda}{\ell}} N - \lambda C
\]

(7)

\[
\frac{dT_f}{dt} = A N - B (T_f - T_a)
\]

(8)

\[
\frac{dT_a}{dt} = B (T_f - T_a) - D (T_a - T_i)
\]

(9)

\[
\rho = -\frac{f}{\delta} (T_a - T_{a0})
\]

(10)

where

- \( N \): Neutron density,
- \( C \): Delayed neutron precursor concentration,
- \( T_f \): Fuel temperature,
- \( T_a \): Average coolant temperature,
- \( T_i \): Inlet coolant temperature, = 50 \degree C (constant),
- \( \rho \): Reactivity,
- \( \beta \): Delayed neutron fraction, = 0.0075,
- \( \ell \): Neutron generation time, = 0.001 sec,
- \( \lambda \): Average decay constant for delayed neutron precursor, = 0.08 sec,
- \( A \): 1, \( B \): 5, \( D \): 2,
- \( T_{a0} \): Mean value of average coolant temperature, = 100 \degree C,
- \( \delta \): Temperature feedback coefficient adopted, as parameter.

If \( f \geq 0.00372 \), this system becomes oscillatory. Variables \( N \) and \( T \) are assumed to be those observable. Following Proposition 1, \( N \) and \( T_a \) are derived digital procedure with sampling time. There being 4 signals generated, 6 patterns \((N, \dot{N}), (T_a, \dot{T}_a), (N, T_a), (T_a, \dot{N}), (N, T_a) \) are obtained.

The patterns shown in Fig. 7 and Fig. 8 were obtained for \( f = 0.002 \), and \( f = 0.00558 \), respectively. Very good correspondence is seen between stochastic and deterministic patterns. Reactivity excitation was assumed in both cases.

Fig. 7. Deterministic and stochastic patterns and time records of reactor system with temperature feedback. (\( f = 0.002 \), S.D. = 7.5 \times 10^{-5}, S.T. = 0.02 sec)
5.2. Reactor system with void feedback

One typical reactor system, with void feedback, is a Boiling Water Reactor (BWR). The deterministic patterns depicted in state space were already studied by Jose March-Leuba, Dan C. Cacuci, and Rafael B. Perez, who found interesting phenomena, that is, bifurcation and chaos in its unstable condition. (March-Leuba, 1986) For stochastic excitation, they showed that BWRs exhibit a single characteristic resonance, at 0.5 Hz, in the linear regime, and by contrast, harmonics of this characteristic frequency appear in the nonlinear regime. Furthermore, their work also demonstrates that amplitudes of the limit cycle oscillations do not depend on the variance of the stochastic excitation and remain bounded at all times.

It is easily deduced, from the authors study on two-dimensional nonlinear system, described by Van der Pol equation, that the stochastic patterns depicted in state space will correspond to the deterministic patterns in that space and the fluctuation around the average trajectory will depend on the variance in the stochastic excitation. First, the authors followed their work with the same model as they used for deterministic pattern and then studied the stochastic pattern. Reactivity excitation was assumed for both cases.

The ordinary differential equation and its parameter of a dynamic BWR system are quoted from their paper:

\[ \frac{dn(t)}{dt} = \left( \frac{\rho - \beta}{A} \right) n(t) + \lambda c(t) + \frac{\rho(t)}{A} \]  \hspace{1cm} (11)

\[ \frac{dc(t)}{dt} = \frac{\beta}{A} n(t) - \lambda c(t) \]  \hspace{1cm} (12)

\[ \frac{dT(t)}{dt} = a_1 n(t) - a_2 T(t) \]  \hspace{1cm} (13)

\[ \frac{d^2 \rho(t)}{dt^2} + a_3 \frac{d\rho(t)}{dt} + a_4 \rho(t) = K T(t) \]  \hspace{1cm} (14)

\[ \rho(t) = \rho_0(t) + D T(t) \]  \hspace{1cm} (15)

where

\( n(t) \): excess neutron density normalized to the steady-state neutron density,
\( c(t) \): excess delayed neutron precursors concentration, also normalized to the steady-state neutron density,
\( T(t) \): excess average fuel temperature,
\( \beta(t) \): excess void reactivity feedback.

Model parameter are shown as below,

\[
\begin{align*}
a_1 & : 25.04 \quad \text{K s}^{-1} \\
\alpha_2 & : 0.23 \quad \text{s}^{-1} \\
\alpha_3 & : 2.25 \quad \text{s}^{-1} \\
\alpha_4 & : 6.82 \quad \text{s}^{-1}
\end{align*}
\]

\[
\begin{align*}
k_0 & : -3.70 \times 10^{-3} \quad \text{K}^{-1} \text{s}^{-2} \\
\delta & : -2.52 \times 10^{-5} \quad \text{K}^{-1} \\
\beta & : 0.0056 \\
\Lambda & : 4.00 \times 10^{-5} \quad \text{s}^{-1}
\end{align*}
\]

which are Model Parameters for Vermont Yankee Test 7N (Sandoz, 1983 and March-Leuba, 1986).

Figure 9 shows the patterns in case of \( k=k_0 \times 0.8 \), that is, stable state. In this BWR case, the stochastic patterns are composed of fast and slow parts. The former is related with \( \ell/\beta \), a neutronically time constant, and the latter a thermo-hydraulic time constant which is much larger than \( \ell/\beta \). As this evidence, when a short time memory is used, the resulting patterns in Fig.10 exhibit only a fast part related with a neutronically parameter. In the case of fuel temperature input, the resulting patterns in Fig.11 show only slow part, as the fast behavior is filtered by thermo-hydraulic characteristics. On the other hand, Figs.12 and 13 show patterns in cases of \( k=k_0 \times 1.2 \) and \( k=k_0 \times 1.55 \), that is, unstable state. As they found the bifurcation, such a phenomenon appears in case of \( k=k \times 1.55 \) and is seen in both deterministic and stochastic patterns depicted in state space. Fluctuation was also observed, depending on the variance in noise, as expected. As far as only temperature feedback model in Section 4 is concerned, there is no appearance of bifurcation and chaos. However, this problem is outside the scope of this study.

---

**Fig.9.** Deterministic and stochastic patterns of BWR with reactivity noise.

\( k=k_0 \times 0.8, \text{S.D.}=3 \times 10^{-4}, \text{S.T.}=0.001 \text{sec} \)
Fig. 10. Stochastic patterns of BWR with short time memory with reactivity noise. $k=k_0*0.8, S.D.=3*10^{-4}, S.T.=0.001$sec)

Fig. 11. Deterministic and stochastic patterns of BWR with temperature noise. $k=k_0*0.8, S.D.=0.6, S.T.=0.2$sec)
Monitoring reactor system by state space trajectory pattern

(a) $k_0=1.2$, S.D. = $2 \times 10^{-5}$, S.T. = 0.001 sec  (b) $k_0=1.55$, S.D. = $2 \times 10^{-5}$, S.T. = 0.001 sec.

Fig. 12. Deterministic and stochastic patterns of BWR with reactivity noise.

(a) Deterministic time record.  (b) Stochastic time record.

Fig. 13. Deterministic and stochastic time record.
($k_0=1.55$, S.D. = $2 \times 10^{-5}$, S.T. = 0.001 sec)

6. CONCLUSION

A method for monitoring the system state by state space trajectory pattern was proposed. Computer simulation was performed on simple systems and reactor systems. A one-to-one correspondence is ensured between a system state and its trajectory pattern in the state space, both for deterministic and stochastic phenomena. This method permits direct treatment of raw data and also nonlinear systems. Signals are always in state space, so long as the system is stable in terms of Liapunov's definition. The increase permitted in the number of dimensions affords commensurate extension of the range of derivable system information. A region affected by abnormal condition can be indicated in either two- or three-dimensional space. This method will serve as a kind of graphic monitoring of reactor noise diagnosis.

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REFERENCES

APPLICATION OF REACTOR NOISE MODELS FOR THE ANALYSIS OF THERMOHYDRAULIC FEEDBACK

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Abstract - In the paper the eigenvalues of a space-dependent coupled neutronic-thermo hydraulic noise model are studied. These eigenvalues are very useful tools of the investigations of thermo hydraulic feedback effects. There is marked difference between the eigenvalues of models with and without feedback. The critical parameter-combinations introduced in the paper help to reveal the origin of this difference. The critical combinations of the coolant velocity and fuel-to-coolant heat transfer coefficient are determined for a PWR core. It is shown, that critical combinations are related to low-frequency resonances and their presence is an inherent feature of light water reactors.

1. INTRODUCTION

The usual way of the reactor noise diagnostics is the analysis of measured spectra. Making use of a core model a method for the calculation noise spectra can be established. Comparing the measured and calculated spectra the correctness of the model can be checked. As a next step the application of the model for diagnostical aims follows. This application means searching for characteristic forms and patterns and also looking for the effects of variation of the most important parameters of the model. The above study is considered to be successful when similar patterns are found in the measured spectra.

The final aim of this approach is to determine the spectral quantities /APSD, coherence, phase and amplitude of CPSD/ of the reactor noise, for also the measured results mostly appear as a spectra.

Besides these an other, nontrivial method exists also in order to solve problems of reactor noise diagnostics. In the theoretical model there are quantities which are not related directly to a measurable quantity or to a combination of them, however they have useful information concerning the behaviour of the core. Further on these quantities will be referred to as derived variables of the model, which provide information not merely on a single physical process, but on the interaction of several processes.

In the course of the investigations the space of the measured quantities are left, and a newly created space of derived variables is used in order to understand better the core processes. The derived variables have direct mathematical meaning and the behaviour of the system can be studied most easily with the help of them. In this way the relationships of crucial importance can be clearly seen. The formulae obtained this way are usually simple and mathematically easily treatable.

A disadvantage of the derived quantities follows from their definition. They do not have trivial physical meaning at the beginning of the investigation. Only further analysis of the model can help to reveal their physical content.

Let us take e.g. the coupled neutronic-thermo hydraulic model. With the use of matrix equations a method was established for the calculation of PSDs, Kozma et.al. /1983/. The spectra are expressed in terms of the eigenvalues of the so-called system-matrix. The comparison of the measured spectra with the calculated ones helped to understand better some practical problems of noise diagnostics, Kozma, Mesko /1985/, Kozma, Katona /1986/. Thus the above-mentioned results belong to the usual direction of noise investigations.
It became clear during the calculations, that the eigenvalues contain useful information on the reactor. Apart from the fact that eigenvalues are used when calculating the spectra, they themselves deserve particular attention. According to the above discussion the eigenvalues can be regarded as derived quantities.

As the physical parameters of the core model are found in the system-matrix, thus analysing the spectrum of this matrix one can follow all the important physical processes in the core. However in practice it is not easy to gain information from the eigenvalues. The eigenvalues of the reactor model based on the equations of the point-kinetics with feedback determine the stability of the reactor, Bell and Glasstone /1970/.

Inclusion of a simple, one-dimensional heat-balance equation into the model of point-kinetics, results in a low-frequency peak in the PSD of neutron noise and makes it possible the monitoring of the thermophysical parameters of the core, Mirsoyjan /1967/. It is not easy to interpret the eigenvalues of a space-dependent neutronic model. If the study is restricted to the axially propagating density fluctuations, then at not very low frequencies /v > 1 Hz/ the obtained eigenvalues can be related to two different spatial relaxation lengths, Behringer et al. /1979/. The first one /λ/ describes a reactor noise component that changes rapidly in space, while the other one /μ/ belongs to the noise component, with relaxation length comparable with the size of the reactor.

The situation becomes more complicated if one takes into account, that the eigenvalues of the above model are in fact complex numbers. This property is especially important in the case of μ at low frequencies, thus the neutron noise component belonging to μ can not be treated below 1 Hz as pure relaxation term.

The physical interpretation of the eigenvalues of a coupled neutronic-thermohydraulic model is even more difficult. As thermohydraulic feedback appears at low frequency, thus we expect new results from this model in that frequency region.

According to earlier studies a really new effect arose in space-dependent coupled models of power reactors at low frequencies, Kozma /1985/. This effect, the so-called low-frequency splitting of eigenvalues is in close connection with the presence of a medium-range phenomenon in power reactors.

In the followings the low-frequency behaviour of the eigenvalues will be studied in detail. It will be shown that an inherent singularity occurs in the coupled models, that can be related to the stability of the reactor.

2. THE SPACE-DEPENDENT COUPLED MODEL

The most important properties of the model will be given here briefly, for details see Kozma /1985/.

The model consists of a neutronic and thermohydraulic part. The neutronic part includes the two-group diffusion equations for the neutron flux with one group of delayed neutrons. For the description of neutronic processes the following state-variables are used: N₁ and N₂ are the fast and thermal neutron fluxes, N₁₀ and N₂₀ are the gradients of the fast and thermal neutron fluxes, respectively. The independent variables are the axial space-coordinate /z/ and the time /t/.

The thermohydraulic part of the model depends on the actual phenomenon to be investigated. For the complete thermohydraulic description the mass-, energy- and momentum-conservation equations of the coolant are used. These equations include three state-variables: T, v and p, the temperature, velocity and pressure of the coolant, respectively. The above variables describe one-phase flow, however the model can be easily extended for two-phase flow, as well. Now we will set up a model for one-phase flow, because already in this simple case the effect to be analysed will be seen well. The coolant is supposed to be incompressible. In this case it is sufficient to take into account only the energy-conservation equation. There are several ways to consider the transfer of heat from the fuel to the coolant. For this purpose we use the heat-balance equation of the fuel with an effective fuel-to-coolant heat transfer coefficient /h/.

The coupling between the neutronic and thermohydraulic parts is created on the one hand by heat getting into the coolant via the coefficient h, on the other hand through the dependence of macroscopic cross-sections on the density and temperature of the coolant.
In this model two essentially different processes are coupled. While neutronic processes are modelled by diffusion equations, the thermohydraulics of the system is determined by the undirected flow of the coolant in the reactor channel. Mathematically these processes are described by second order partial differential equations in the first case, and by first order partial differential equations in the second case. Consequently, the efficient methods developed in order to to solve system of diffusion equations are not applicable for the coupled model.

Performing Fourier-transformation with regard to the time variable the coupled system of partial differential equations can be written in the form

\[
\frac{\partial x}{\partial z} + A x = f, \quad /2.1/
\]

where \( x = x(z, \omega) \) - state vector, which consists of the space- and frequency-dependent neutronic and thermohydraulic state-variables,

\( A \) - system-matrix containing all the physical parameters of the reactor model,

\( f \) - vector of noise sources in the core.

Using the necessary boundary conditions and the Wiener-Khinchin theorem, the matrix of the power spectral density functions /PSD/ can be calculated. The /i,j/-th element of the PSD matrix is the CPSD between the i-th and j-th state-variables in the state-vector. In the calculation of the PSD matrix the eigenvalues of the system-matrix play an important role.

On the basis of these calculations the model was applied for the analysis of practical problems. First the low-frequency behaviour of temperature and neutron CPSDs was studied by the model Kozma, Meskő /1985/. The slope of the phase of CPSDs between in-core neutron detectors and core-exit thermocouples was analysed in Kozma, Katona /1986/. Periodical minima /sinks/ of the APSDs and CPSDs of neutron signals are treated in the work of Katona, Kozma /1987/.

Further on not the presentation of the results of PSD calculations will be continued, but the eigenvalues of the system-matrix will be studied.

3. THE DEGENERACY OF THE SYSTEM-MATRIX

The system-matrix is defined by the Eq. /2.1/, and has the following form:

\[
\begin{pmatrix}
0 & 0 & 0 & 0 & 0 & 0 & -1/D_2 \\
0 & 0 & 0 & 0 & 0 & 0 & -1/D_2 \\
0 & 0 & 0 & 0 & -1/\mu & 0 & 0 \\
0 & \lambda_3 & \lambda_4 & i\omega & 0 & 0 & 0 \\
0 & \lambda_3/\lambda_4 & (\lambda_3+i\omega)/\lambda_4 & \lambda_3/\lambda_4 & 0 & 0 & 0 \\
-\lambda & \lambda & \lambda & \lambda & \lambda & \lambda & \lambda \\
-\lambda & \lambda & \lambda & \lambda & \lambda & \lambda & \lambda
\end{pmatrix} /3.1/
\]

The notations are:

\( D_1, D_2 \) - fast and thermal diffusion coefficients,
\( \Sigma_1, \Sigma_2 \) - fast and thermal macroscopic cross-sections,
\( \Sigma_R, \Sigma_f \) - removal and fission macroscopic cross-sections,
\( v_1, v_2 \) - velocity of fast and thermal neutrons,
\( \theta(u) \) - a function taking into account the effect of delayed neutrons,
\( \lambda_{ij} \) - various coupling coefficients,
\( v \) - the velocity of the coolant in the reactor channel.

In the /3.1/ expression the frequency appears, thus the eigenvalues \( \kappa \) of \( \Lambda \) also depend on the frequency, \( \kappa = \kappa(u) \).
In the numerical model the data of a 1400 MWth pressurized water reactor are used. The 7x7 matrix \( \Lambda \) generally has 7 eigenvalues. 4 eigenvalues are practically independent on the frequency. They are denoted by \( \kappa_1, \kappa_2, \kappa_3, \) and \( \kappa_4 \). The remained 3 ones, \( \kappa_5, \kappa_6 \) and \( \kappa_7 \) depend on the frequency. \( \kappa_1 \) and \( \kappa_2 \) and also \( \kappa_3 \) and \( \kappa_4 \) form a pair. For the first pair holds:
\[
\kappa_1 = - \kappa_2 = - \frac{1}{L},
\]
where \( L \) is the diffusion length of neutrons. It is clear from Eq. /3.2/ that \( \kappa_1 \) corresponds to the \( \lambda \) eigenvalue introduced by Behringer et. al. /1979/. The other pair of eigenvalues writes
\[
\kappa_3 = - \kappa_4 = K,
\]
where \( K \) is a constant that depends on the thermohydraulic parameters of the core, and is more than an order larger than \( 1/L \). The eigenvalues are calculated in cm\(^{-1}\). In the PSD calculations all the 7 eigenvalues are used, however now we concentrate on the 3 frequency dependent ones.

On Fig.1. \( \kappa_5, \kappa_6 \) and \( \kappa_7 \) are depicted. The parameter of the curves is the frequency. Circle markings correspond to the increasing frequency values in log scale from 0.001 Hz to 10 Hz.

On Fig.1. the eigenvalues of a model with normal operating parameters are shown, on this figure the thermohydraulic feedback is taken into account. On Fig.2. the eigenvalues of a model without feedback are depicted. Neglecting feedback means on the one hand, that the effect of neutron noise field on thermohydraulic fluctuations is disregarded. On the other hand the effect of thermohydraulic processes on the neutronics is still taken into account.

This simplification yields the same model as the transfer-function approach, which is a frequently used tool of the reactor noise theory. In that approach the response of the reactor to a thermohydraulic perturbation is evaluated with the help of a reactor transfer function. This transfer function is determined on the basis of a neutronic model of the core, e.g. point kinetics, diffusion model etc.

It is not difficult to obtain curves of Fig.2. from the coupled model, simply the substitution \( h=0 \) has to be done.

The eigenvalues of the case without feedback are \( \kappa_i^0 \) \( i = 1, ..., 7 \). At high frequencies \( \omega \gg 1 \) Hz/ \( \kappa_5 \) and \( \kappa_6 \) do not depend on the coupling, but \( \kappa_7 \) depends on it. \( \kappa_7 \) draws near to the imaginary axis, but does not reach it in the investigated range of the frequency parameter. One may observe, that \( \kappa_7^0 \) coincides with the positive half of the imaginary axis on the Fig.2.

It is also seen on Fig.1. and Fig.2. that at low frequencies the curves with and without feedback split and have different shapes depending on the actual value of \( h \). This phenomenon is called the low-frequency splitting, Kozma /1985/. The curve of \( \kappa_7 \) on Fig.1. can be divided into two characteristic parts. At low frequencies it behaves similarly as the low-frequency part of \( \kappa_5^0 \), however at higher frequencies the picture changes and \( \kappa_7 \) approaches again.
For very small values of \( h \) the curves of the eigenvalues may not differ too much from the curves of Fig. 2. Thus there has to be a certain value of \( h_{\text{cr}} \) /critical value of \( h \)/ with the following properties. If close to \( h_{\text{cr}} \) and \( h < h_{\text{cr}} \), then both the low- and high-frequency part of \( \kappa_7 \) behaves like \( \kappa_7^0 \). If \( h \) close to \( h_{\text{cr}} \) and \( h > h_{\text{cr}} \), then the low-frequency part of \( \kappa_7 \) similar to \( \kappa_5^0 \), and only the high frequency part behaves like \( \kappa_7^0 \).

The above conclusion is valid also in the case of fixed \( h \) but varying \( v \), thus there exists a critical velocity value for every fixed \( h \). The only change is that velocity has the opposite effect on the eigenvalues as \( h \). In other words high velocities yield curves with behaviour shown on Fig. 2. /there is no feedback/, and low frequencies correspond to strong feedback /Fig.1./. It is possible to select an other parameter of the model, e.g. the power level of the reactor, and to define the critical value of this quantity /\( P_{\text{cr}} \)/.

Let us generalize these results and select \( k \) variables from the physical parameters of the model. These variables form a \( k \)-dimensional space. Fixing the values of /\( k-1 \)/ parameters the remained one has /at least/ one critical value. In some left and right neighbourhood of this critical value the behaviour of this eigenvalue is essentially different. Thus the critical combinations of parameters form a hypersurface of co-dimension one in the space of dimension \( k \).

Easy to see that in the case of critical parameter values the curves belonging to \( \kappa_5 \) and \( \kappa_7 \) cut each other such that in the point of intersection they have the same value of frequency /\( \omega_\text{cr} \)/. In other words, in the case of critical parameter combinations there is a certain frequency /\( \omega_\text{cr} \)/ at which two of the eigenvalues coincide.
So the above hypersurface of co-dimension one is the hypersurface of degeneracy of the system-matrix.

Let us consider e.g. the 3-dimensional space of variables $h$, $v$ and $P_0$. Then the critical parameter-combinations form a surface in 3-space. Fixing the parameter $P_0$, the points of the critical $h,v$ combinations form a curve in the plane $P_0=\text{const}$.

Let us take the flow velocity value $3.66 \text{ m/s}$. This velocity forms a critical pair with heat transfer coefficient $h=0.92 \text{ W/cm}^2\text{°C}$. It is illustrated on Fig. 3, how the eigenvalue curves vary with increasing $h$ in the neighbourhood of $h_{cr}$. The $\omega_{cr}$ value is 0.17 Hz at that $h_{cr}$. In the neighbourhood of $h_{cr}$ the curves of the eigenvalues are close to each other, excluding only the sections with frequencies in the neighbourhood of $\omega_{cr}$. The intersection of the curves is presented enlarged, where one can follow how the curves change when $h$ varies.

At the intersection of the curves we have saddle point, which is well-known from the theory of nonlinear differential equations, Arnold /1980/.

Different velocities give different values of $h_{cr}$, see Fig. 4, and Fig. 5. Curves on Fig. 4, are calculated with $v=5 \text{ m/s}$, the value of $h_{cr}$ is equal to 1.62 $\text{ W/cm}^2\text{°C}$. The intersection of the curves is again of saddle point type. The critical frequency value changes but little and equals to 0.22 Hz.

For Fig. 5, a relatively low velocity value was chosen, $v=2 \text{ m/s}$, which yields $h_{cr} = 0.06 \text{ W/cm}^2\text{°C}$ and $\omega_{cr} = 0.33 \text{ Hz}$. Now the picture changes more significantly and the intersection is a deformed saddle point. Due to the low $h$ value the eigenvalues are more sensitive to the variation of this coefficient.

4. SINGULARITIES ON THE HYPERSURFACE OF DEGENERACY
(DISTINGUISHED CRITICAL PARAMETER-COMBINATIONS)

The starting point of the investigations in the present chapter is the difference between Fig. 4, and Fig. 5. Evaluation of critical combinations for further velocity values results Fig. 6, where the $h_{cr}(v)$ function is shown in the velocity range of 0.5 to 5 m/s. This function consists of two linear parts for low and high velocity values, respectively.

The first part extends to approx. 2 m/s and has small slope, the second part begins at approx. 2.5 m/s and it has a higher slope. Between these two linear parts there is a transitional section, where $h_{cr}$ changes rapidly in a small range of velocity. If the velocity fluctuates around 2.25 m/s and the heat transfer coefficient is less than 0.5 $\text{ W/cm}^2\text{°C}$, then critical combinations can easily arise.

It is not easy to estimate $h$ of a given reactor. Simple thermophysical considerations result fuel time constant $\sim 10 \text{ s}$, Katona /1985/. However measurements are not in accordance with such a high time constant, and give rise to smaller values, Kleiss, van Dam /1985/.
Fig. 4. The eigenvalues of the case with \( v = 5 \text{ m/s} \). For the first solid lines \( h = 1.65 \text{ W/cm}^2\text{°C} \), for dotted lines \( h = 1.6 \text{ W/cm}^2\text{°C} \). The critical frequency value is 0.22 Hz.

Dividing the fuel pellet into cylindrical regions yields better agreement between the calculations and the measurements, Hagen /1986/.

Fig. 5. The eigenvalues of the case with \( v = 2 \text{ m/s} \). For the first solid lines \( h = 0.08 \text{ W/cm}^2\text{°C} \), for dotted lines \( h = 0.03 \text{ W/cm}^2\text{°C} \). The critical frequency value is 0.33 Hz.

The relationship between the time constant \( \tau \) and the heat capacity \( h \) in the case of cylindrical geometry writes:

\[
h = \frac{c_F h_F}{\tau}
\]

Thus for a given time constant \( \tau \) can be calculated. Here:

- \( c_F = c_{\text{PP}} \rho_F \) — product of the specific heat and the density of the fuel \((\text{J/cm}^3\text{°C})\)
- \( R_F \) — radius of the fuel pin \((\text{cm})\)
- \( \tau \) — fuel time constant \((\text{s})\).

Substituting the necessary parameter values into /4.1/ we obtain, that in LWRs for \( h \) holds:

\[0.1 \text{ W/cm}^2\text{°C} \leq h \leq 1 \text{ W/cm}^2\text{°C}.

Thus it cannot be excluded, that a parameter-combination with \( v \) fluctuating around 2.25 m/s is found in the transitional section of Fig. 6.

The presence of the transition period on Fig. 6. shows that the hypersurface of degeneracy has its own singularities e.g. at 2.25 m/s. These singularities must be avoided even in case of short time transients.

This singularity appears on Fig. 7., too, where the dependence of \( \omega_{\text{cr}} \) on \( v \) is depicted. There is two linear parts on the figure and a transitional section between them. The critical frequency value is found between 0.15 Hz and 0.4 Hz.
Fig. 6. The critical fuel-to-coolant heat transfer coefficient as a function of flow velocity.

Fig. 7. The critical frequency value as a function of flow velocity.

Fig. 7. confirms that \( v = 2.25 \) m/s is a distinguished value, because in the region of this velocity there is a break in the \( \omega_{cr}(v) \) curve. This part of the figure is drawn by dotted line.

Further properties of the hypersurface of degeneracy can be revealed by fixing a group of parameters and varying some others. The singularities of this hypersurface deserve particular attention when applying the above theory for practical aims. In the applications one has to consider that the singularities are in fact low-frequency resonances. The properties of these resonances depend on the physical parameters of the core (e.g. \( h, v, P_0 \)). That allows to monitor these parameters via the analysis of the low-frequency resonances.

5. CONCLUSIONS

The eigenvalues of the system-matrix of a coupled neutronic-thermohydraulic reactor noise model can be studied with the help of simple mathematical methods, contrary to the complicated matrix-functions used during the calculations of PSDS. The physical meaning of the eigenvalues is not trivial and it can be revealed only during the further analysis.

It is shown that two of the eigenvalues of the system-matrix are coincide at certain parameter values. This phenomenon is the degeneracy of the system-matrix. In our case the degeneracy is due to the fact that two essentially different processes, the diffusion and the convection, are coupled in the model. The degeneracy of the system-matrix is characterised by critical parameter-combinations.

The critical velocity and fuel-to-coolant heat transfer coefficient combinations were determined assuming the remained core parameters to be constant. Every critical parameter-combination has its own critical frequency value, at which the degeneracy takes place.

The critical parameter-combinations form a hypersurface in the space of the investigated parameters. It was found that this hypersurface is not smooth, and has its own singularities. A singular velocity value is \( \sim 2.25 \) m/s. This velocity is characterised by that special property, that in case of small velocity variations the critical heat transfer coefficient varies significantly.

In the present investigations only 2 parameters (\( v \) and \( h \)) were studied for the sake of simplicity. It is very important to investigate also the effect of varying power level, because that helps to describe in detail the singularities of the hypersurface of degeneracy.

In the applications the space of derived quantities is to be left. The singularities yield low-frequency resonances. For the analysis of these resonances no simple analytical relationship exists as for Mirsoyan /1967/. Nevertheless, the analysis can be performed with the help of the method described in Chapter 2, but
this will be the goal of a forthcoming paper.

Acknowledgement - The author wishes to express his gratitude to Dr. Thie for calling his attention to the low-frequency thermohydraulic resonances.

REFERENCES
MEASUREMENT METHODS

Session chairman: R. Albrecht (U.S.A.)
SUMMARY OF THE SESSION

Eight papers are included in this session. Four of the papers deal with noise analysis methods applied to the understanding of fuel performance. The other four papers represent a variety of other applications. These include the estimation of control rod effectiveness during xenon transients, the early detection of local boiling, the investigation of two phase flow properties by X-ray tomography and the observation of space dependent stochastic fluctuations in a large graphite reactor.

Morishima and Turkcan present a simple index to express the control-rod effectiveness, the covariance to variance ratio. This ratio is easy to estimate and it has a clear physical meaning. However, care must be taken to remove the almost deterministic component due to slow feedback effects by trend removal.

The paper by Hummel and Wesser shows spatial distributions of various moments of data obtained from an air-water two phase flow loop. Spatially dependent spectra are also presented. The results show promise for two-phase flow investigations aimed at basic understanding. However, it is clear that much more refined analyses are required for these X-ray tomography techniques to make a significant contribution to the fundamental understanding of two-phase flow phenomena.

Deconinck and Bouneder shows the results of an experimental investigation of the detection of the localized onset of boiling in water on a heated fuel pin. Accelerometers placed on the pin and on the surrounding structures at appreciable distance from the boiling position were used. The technique proved to be very sensitive because of the transmission of mechanical energy from the boiling to the mechanical environment.

Oguma et al. reported on a recursive identification technique based on an adaptive lattice filter. Results were shown for fuel thermal time constant estimation and for the mechanical response (elongation) under steady and transient conditions. The recursive identification technique is useful for studying fuel behaviour during slowly varying state changes.

Crowe et al. reports on new measurements made in a large graphite reactor. Low frequency phase shifts between axially displaced neutron detectors were observed and rationalized by a comparison to calculation. At higher frequencies, the behaviour of phases between various detectors was not supported by computations. Also, an unexplained 1.4 Hz small resonance was observed in this reactor. The results showed that future measurements with a fully deployed array of incore sensors in N-Reactor are likely to lead to a much more complete and comprehensive understanding of these large systems.

The paper by Edelmann et al. use the same data from Super Phenix I. The analysis in this paper concentrated on the frequency range between 0.001 Hz and 1.0 Hz. The result is relatively pessimistic for noise analysis. The major part of temperature noise measured above the Super Phenix I subassembly outlets is not related to fluctuation of reactor or subassembly power.

The three remaining papers on fuel performance all deal with evaluating LMFBR fuel. Girard et al. report on the analysis of good quality temperature noise signals obtained during the commissioning phase of Super Phenix I. The signals from 448 instrumented subassemblies were analyzed for r.m.s levels and other statistical parameters. The data came from a combination composite thermocouple and intrinsic thermocouple. The data allowed the determination of the k-value (normalized r.m.s temperature) which was shown to remain close to a constant as reactor power and mass flow vary.
APPLICATION OF NOISE ANALYSIS FOR ESTIMATION OF CONTROL ROD EFFECTIVENESS IN THE HFR PETTEN AND THE BORSSELE PWR DURING THE XENON TRANSIENT

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ABSTRACT - The reactivity effectiveness of a control rod is estimated for the high flux reactor in Petten at automatic full-power operation shortly after startup. Transient noise processes arising largely from fission-product poisoning are fully utilized to arrive at estimates of the control-rod effectiveness defined as the covariance-to-variance ratio for neutron noise and fine-rod's successive movements. The physical meaning of the effectiveness is explained in terms of a reactivity balance associated with control and feedback reactivity. A time-domain noise analysis for many small pieces of transient data is performed to obtain useful information about the time-varying properties of the system such as, for example, a gradual change in fine-reactivity adjustments from frequent regulation to smoother control. Application to the Borssele PWR has also been made to show steady effectiveness of a control rod bank during 13 hours after start-up at full-power.

KEYWORDS
Transient noise data, reactor noise analysis, fission product poisoning, high flux reactor, automatic control, reactivity balance, covariance-to-variance ratio, fine-rod movement, PWR noise, trend estimation

1. INTRODUCTION

The reactivity effectiveness of a control rod is very important in operating a reactor at fixed power level for a predetermined period of time. Usually, it is defined in terms of reactivity worth, total and/or differential, in the core configuration in which reactivity change takes place during power operation due to processes such as fuel burnup, fission-product poisoning and reactor control operation. A simple diagram of a reactivity balance is shown in Fig. 1. The total reactivity $R_t$ is given by

$$R_t = R_c + R_f + R_d$$

(1)

where $R_c$ is the reactivity worth of all control rods, $R_f$ the feedback reactivity at power operation, and $R_d$ the reactivity disturbances from various random processes. On a time scale greater than hours, the total reactivity is kept at zero for constant power operation. On a much shorter time scale, it always fluctuates around zero, owing largely to $R_d$ and also $R_c$ for fine regulation.

In the present paper, the neutron noise and control-rod movement on the shorter time scale are utilized for estimation of the reactivity effectiveness $R_t$ of a control rod at operation. For this, a simple and useful index is proposed, namely to express $R_e$ as the ratio of the covariance between relative power fluctuations and control-rod movement, to the variance of control-rod movement. The meaning of $R_e$ is interpreted physically in terms of the reactivity balance based on the over-all transfer function from control rod to neu-
tron flux, which is described in detail in the next chapter.

On the basis of the physical meaning of $R_e$, noise measurements and analysis have been made on the High Flux materials testing Reactor (HFR) of the Joint Research Centre Petten (Röttger et al., 1986) and also on the Borssele PWR (Türkcan, 1982) at full power shortly after start-up. The results on the HFR are discussed in detail in view of the reactivity balance and the time-varying behavior of fine-reactivity adjustments. As for the Borssele PWR, the results of $R_e$ are treated in connection with other reactivity feedbacks such as boron concentration and thermo-hydraulic processes. It is also shown that a time-domain noise analysis for many small pieces of data is useful for an understanding of the control rod effectiveness, for example, its gradual decrease with operation time.

Until now, flow-induced mechanical vibrations of control rods have been studied in many different reactors with the purpose of monitoring and detecting incipient signs of failure as soon as possible (e.g., Fry, 1971; Shono et al., 1985; Mingchang et al., 1985; Mullens et al., 1985). On the other hand, by the use of the signal to control the rod movement for reactivity regulation, the determination of reactor control system dynamics has been attempted in the 2-MW research reactor HOR at Delft (Hoogenboom et al., 1982). The present study puts emphasis on the estimation of the reactivity effectiveness of a control rod during automatic operation.

2. PHYSICAL MEANING OF THE CONTROL-ROD EFFECTIVENESS

The neutron-flux level is always regulated automatically and/or manually through a reactor control system, resulting in fluctuations around a certain level, under the presence of some inherent noise sources, both neutronic and thermo-hydraulic, especially on a time scale much shorter than hours. In the beginning of full-power operation shortly after start-up, such a regulation is highly significant, because there is a transient reactivity change largely due to fission-product poisoning. Hence a control rod (or member) is frequently moved upward and downward as if it fluctuates having a marked correlation with the neutron-flux fluctuations.

The present noise analysis utilizes both the fluctuations in neutron flux and control-rod movement in order to determine the degree of the correlation. The following estimates are first obtained.

\[
\begin{align*}
V_{nn} &= \text{Variance of neutron noise, expressed in (\% of full-power)}, \\
V_{cc} &= \text{Variance of control-rod movement in cm}, \\
V_{cn} &= \text{Covariance between them in \%,cm}. \\
\end{align*}
\]

As a simple index to express the control-rod effectiveness, a covariance-to-variance ratio $R_e$ is defined:

\[
R_e = \frac{V_{cn}}{V_{cc}} \text{ in \%/cm}
\]  \hspace{1cm} (2)

where

\[
-\sqrt{\frac{V_{nn}}{V_{cc}}} \leq R_e \leq \sqrt{\frac{V_{nn}}{V_{cc}}}
\]  \hspace{1cm} (3)

in view of the limits for the cross-correlation coefficient $V_{cn}/\sqrt{V_{nn}V_{cc}}$ ranging from -1 to 1.

The ratio can be interpreted physically as follows. Expressing $V_{nn}$ and $V_{cc}$ by the relevant cross-PSD $P_{cn}(f)$ and auto-PSD $P_{cc}(f)$, respectively, where $f$ is the frequency in Hz, we obtain

\[
R_e = \frac{\int_{-\infty}^{+\infty} P_{cn}(f) df}{\sqrt{\int_{-\infty}^{+\infty} P_{cc}(f) df}}
\]  \hspace{1cm} (4)
Replacing the $P_\text{cc}(f)$ in Eq. (4) by $P'_\text{cc}(f) \cdot T_{cn}(f)$ where $T_{cn}(f)$ is the over-all (closed-loop) transfer function, we have

$$R_e = \int P'_\text{cc}(f) \cdot T_{cn}(f) \, df$$

(5)

where $P'_\text{cc}(f)$ is the auto-PSD normalized to unit variance in the sense that

$$P'_\text{cc}(f) = P_{cc}(f) \cdot \int P_{cc}(f) \, df.$$  

(6)

The integrand on the right-hand side of Eq. (5) can be expressed in terms of the Fourier integral of the convolution product of $C_{cc}(t)$ by $H_{cn}(t)$ where $C_{cc}(t)$ is the auto-covariance function with the initial value $C_{cc}(0) = 1$ and $H_{cn}(t)$ the impulse response function related to $T_{cn}(f)$. Performing the integration with respect to $f$, then we obtain

$$R_e = \int C_{cc}(-t) \cdot H_{cn}(t) \, dt$$

(7)

Consequently, the ratio $R_e$ indicates the total response of neutron flux being subject to the successive movements of the control rod, i.e. by the $C_{cc}(t)$, through the whole reactor-core dynamics $H_{cn}(t)$, both neutronic and thermo-hydraulic.

It may be worth mentioning that the ratio $R_e$ can be understood directly from the viewpoint of the reactivity balance, especially when the following knowledge is available: (a) The differential reactivity worth $W$ of the control rod is not changed largely during operation: about 59 pcm/cm of $W$ in the HFR has been observed previously by a reactor period measurement. And (b) The neutron flux fluctuations are affected by the successive fine control-rod movement effectively during the past 100 s. In our experiments the slow neutron flux responses are eliminated by a trend removal (see below). This means that $T_{cn}(f)$ is significant at high frequencies above about 0.01 Hz, thus permitting the approximation that the reactivity-to-neutron direct transfer function (i.e. a zero-power reactor transfer function) may be expressed by a constant gain $G$ of about $1/\beta_{\text{eff}} = 0.13\%$/pcm in the frequency range of about 0.01 to 10 Hz.

By the use of the above knowledge, the ratio $R_e$ can be expressed as follows.

$$R_e = W \cdot R_G \cdot G,$$

(8)

where $R_G$ is the covariance-to-variance ratio expressing the effectiveness of successive reactivity regulation $R_c$ on the total reactivity $R_T$, defined by

$$R_G = (\text{Covariance between } R_c \text{ and } R_T) / (\text{Variance of } R_c)$$

(9)

If there is no reactivity feedback and disturbance, the value of $R_e$ must be 1 since $R_c = R_T$ at any moment. Usually it takes a smaller and positive value because some negative feedbacks (i.e. $R_e < 0$) are always present to suppress a reactivity deviation from the original value partly.

3. MEASUREMENTS AND DATA ACQUISITION

Signal measurements have been made in the HFR in Petten during a period of one day shortly after reactor start-up with automatic operation at the full power level of 95 MW. Three kinds of signals have been selected: neutron flux
from one of the safety channels, the position of five coarse rods moving all together in steps, and the position of a fine rod being driven automatically. Figure 2 shows the core configuration of the HFR in which the fine rod is in the position D-6 and the neutron detector used here is indicated by ND#5.

Figure 3 shows the data acquisition system available in the HFR and the data processing system in the noise laboratory of ECN (Türkcan, 1987). In the data acquisition system, the analog signals are separated into two components called DC-part and AC-part, and then digitized at a sampling period of 1 s, which is sufficiently high in view of the fine-rod's regulative movement. The DC-part is taken directly from the analog signal without any filtering and amplification. The AC-part is the noise component resulting from DC-compensation by highpass filtering and amplification. Both amplitudes in the DC- and AC-parts are quantified in 14-bit accuracy between -5 and +5 V. The digitized data are stored on a magnetic tape during the whole period of measurements.

Figure 4 shows the time history of the data for the respective DC-parts. The coarse rods move stepwise only at shimming operation and the fine rod is driven frequently during reactor operation. Both types of control rods move over a small range of about 7 cm along the fuel section of 60 cm length. It is to be noted that even at the end of an operation cycle (26 days after start-up), the control rods are situated only about 13 cm above the initial position. This is due to the high reactivity worth of the control rods, because all of them consist of a cadmium section on top of a fuel section with the same structure and composition. When a control rod is withdrawn, the new fuel moves into the core displacing the cadmium section.

A piece of the AC-part of the signals is shown in Fig. 5. It can be seen that the neutron flux is regulated around a certain level by fine-control adjustments during and after shimming operation. The similar behavior can also be seen from many other pieces of the data, most of which have been utilized for the present noise analysis.

4. CONTROL-ROD EFFECTIVENESS IN THE HFR

The elementary process of neutron noise being subject to reactivity control can easily be understood from the time history in Fig. 5. Except for the period of shimming operation, the fine rod is withdrawn in steps of 0.042 cm on an average, which corresponds to 2.5 pcm in reactivity worth and 0.4% of rise in neutron flux. Then, owing to the gradual xenon poisoning at a rate of 0.022 pcm/s, such a reactivity gain is spent in about 2 minutes, after which the next step of fine-rod withdrawal is started. During the period of 56 minutes between two shimming operations, the fine rod is intermittently driven about 27 times, i.e. at an average rate of 0.008 Hz. Note that use is made of the values of W and G for the above conversions from 0.042 cm to 2.5 pcm and from 2.5 pcm to 0.40 %, respectively, since the coherence between neutron flux and fine-rod movement is highly significant in a narrow frequency range around 0.01-0.1 Hz as will be shown in Fig. 10.

Much the same may be said of the other pieces of the data. To see this actually, the variance and covariance analysis has been made of twenty pieces of the data for neutron noise and fine-rod movement. Every piece is taken from the AC-part data between two successive shimming operations, and preprocessed by the trend removal of second-order polynomial components in both signals. Other types of trend such as a straight line and polynomials of order higher than two were found less satisfactory on account of some undesirable deviations from the data. In the Appendix A, the gradual control-rod movement is explained in terms of xenon poisoning and reactivity addition with the worth of W = 55 pcm/cm.

Figure 6 shows the results of the variance and covariance analysis with respect of the following estimates: the standard deviations of the signals, the cross-correlation coefficients and the reactivity effectiveness R, at
Estimation of control rod effectiveness

20 different times of reactor operation until 24 hours after start-up. Substitution of the mean values of the variances and covariances in the equations (2) and (8), using the given value of G and the W-value from the xenon-poisoning analysis, leads to the estimates \( R_e = 2.7 \times \text{cm} \) and \( R_r = 0.35 \).

Two major features can be observed: (1) The fine-reactivity adjustments are made steadily during automatic operation, which is characterized by the average values of \( R_e \) around 2.7 %/cm and \( R_r \) around 0.35. Namely, in the sense of a stationary process, the control-rod movement of 0.026 cm (i.e., 1.4 pcm) produces a total-reactivity change of 0.49 pcm and then a neutron noise of 0.064 %. Note that this magnitude of neutron noise forms 71 per cent of the standard deviation of about 0.09 %. The difference of 0.026 % arises from some reactivity disturbances due to other noise sources, both neutronic and thermo-hydraulic.

(2) The fine-reactivity adjustments are slightly decreasing with the lapse of reactor operation. This is seen from a gradual decrease in all the estimates in Fig. 6, except for the beginning of full-power operation. To examine on this point, auto-covariance functions of neutron noise have been estimated for pieces of the data analyzed above, but only for those pieces of relatively larger size. Then all of these estimates are fitted to a second-order system model \( C(t) \) to determine two parameters, a time constant \( T \) and a resonant frequency \( F \):\n
\[ C(t) = \exp\left(-\frac{t}{T}\right) \cos(2\pi F t) \]  

(10)

In order to see the period of first decay of \( C(t) \), another parameter \( U \) is calculated by the condition (Morishima, 1987) that

\[ C(U) = \exp(-1.) \text{ for the first time.} \]  

(11)

Figure 7 shows the results on \( T, F \) and \( U \) during the 29 hours of operation after start-up. There is a moderate tendency from a frequent reactivity regulation to a smoother control. This is characterized by a change from damped oscillation into simple decay. The transitional stage is not displayed so well owing to the simple model \( C(t) \) for fitting. A clearer indication of such a transition is shown in Figs. 8 and 9 in terms of auto-PSDs for neutron noise and fine-rod movement, and also in Fig. 10 by the relevant coherence functions. As operation time goes, both the PSDs become more smooth but the coherence is steadily kept in the frequency range of about 0.01 to 0.1 Hz.

5. APPLICATION TO BORSELE PWR

In order to estimate the reactivity effectiveness \( R_e \) of a control rod bank at operation in the Borsele PWR, a noise analysis is made of signals for neutron noise, D- and L-bank movements. The signals have been measured during the period of xenon transient shortly after start-up and after arrival at full-power level. All the settings of measurements such as signal selection, low and high-pass filtering, and AD conversion are manipulated from the ECN laboratory by the on-line real-time data collection system that is connected to the Borsele reactor through a special 200 km long telephone line (Turkcan, 1985). The reactor signals are digitized at a rate of 1 Hz, sent to ECN at Petten, and stored on a magnetic tape for noise analysis of the digital data.

Figure 11 shows a typical part of the data. The neutron flux signal has a series of sharp peaks that come mainly from L-bank movements, though it is not so clear on account of quantization errors. The present analysis utilizes a piece of data on neutron noise and D-bank movement that has a gradual change in trend between two neighboring sharp peaks of neutron flux. Two typical pieces are shown in Fig. 11 by dashed lines. In all, seven pieces of data have been selected and served for estimation of the D-bank effectiveness. The procedure for this is the same as in the case of HFR noise analysis.
Figure 12 shows the performance of reactivity control by the D-bank in terms of the following estimates: standard deviations of neutron noise and D-bank movement, cross-correlation coefficient, and reactivity effectiveness $R_c$ at seven different times of reactor operation until 13 hours after startup. Major results are as follows.

- Neutron flux is steadily regulated within the standard deviation of 0.022% of power on an average during the period of 13 hours. Such a magnitude of standard deviation is about one fifth of that obtained from the whole data including many peaks due to shimming operation (Türkcan, 1985).
- For this, the D-bank is driven frequently just after startup and more gradually later, which is indicated clearly by the gradual decrease in cross-correlation coefficient.
- The reactivity effectiveness $R_c$ is kept almost constant around 0.012% power step as an average over all pieces of data. Hence, 0.79 step of the D-bank movement on an average gives direct rise to 0.0095 %power of neutron noise, i.e., 43% of the standard deviation. Probably, this lower percentage than 71% per cent in the HFR is fully compensated by some other reactivity noise sources. Note that the value of $R_c$ cannot be determined since no data on $W$ are available at this time.

The above findings have also been examined by spectral analysis of the signals on neutron noise and D-bank movement. The same pieces of data as before have been analyzed by using an autoregressive-model fitting method based on Yule-Walker algorithm. The auto- and cross-PSD's have been obtained at the seven different times of operation, of which only the coherence functions are shown in Fig. 13. From the spectral results, it is found that:

- Neutron flux is largely subject to reactivity regulation with D-bank, directly at lower frequencies below 0.02 Hz, because first-order behavior of $f^{-2}$ is in common with both signals and the coherence is significant during the period of 13 hours.
- The magnitude of the coherence function at the lower frequencies decreases with time, especially several hours after startup. This suggests that reactivity regulation is gradually shifting to by chemical shif with boron concentration that has been known to change at regular intervals of about 52.2 minutes.

6. CONCLUDING REMARKS

By the use of noise analysis methods, the reactivity effectiveness of a control rod at automatic operation has been estimated. The covariance-to-variance ratio $R_c$ as well as $R_p$ is easy to estimate and has a clear physical meaning, though care must be taken of the trend removal to subtract the almost deterministic component due to slow feedback effects. From the results for the HFR and Borssele PWR, it is suggested that both ratios are useful in monitoring and evaluating the performance of a reactor control system for many different types of reactors, not only shortly after start-up at full-power operation, but also just after a power-level change from one to another as well as in the case of a load-follow operation.

7. ACKNOWLEDGEMENT

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Fig. 1. Reactivity balance in a reactor at operation.

Fig. 2. Standard core configuration in the HFR. The fine rod is in the position D6 and the neutron detector ND#5 is used for the present measurements.

Fig. 3. Data acquisition system in the HFR and data processing system in the ECN.
Fig. 4. Time history of the DC-part data.

Fig. 5. Time history of a piece of the AC-part data.
Fig. 6. Performance of fine-rod regulation

Fig. 8. Auto-PSDs of neutron noise.

Fig. 7. A time sequence of the parameters characterizing neutron noise under automatic control.

Fig. 9. Auto-PSDs of fine-rod movement.
Fig. 10. Ordinary coherence functions between fine-rod movement and neutron noise.

Fig. 11. Time history of the piece of data on neutron flux, D- and L-bank movements in the Borssele PWR.

Fig. 12. Performance of D-bank regulation by:
- neutron noise standard deviation,
- D-bank movement standard deviation,
- cross-correlation coefficient, and
- the D-bank effectiveness Re.

Fig. 13. Coherence functions between neutron noise and D-bank movement at seven different times of reactor operation.
Appendix A: Determination of Differential Worth $W$ in the HFR

Fifteen pieces of the fine-rod signal data in the DC-part have been utilized for estimation of their upward trends. A second-order polynomial of operation time has been fitted to each piece of the data. The results of the trend estimates are shown in Fig. A-1. It can be seen that the fine rod is withdrawn at a lower rate with increasing operation time. This is due to a gradual saturation of reactivity loss by fission-product poisoning. The major feature of such a poisoning effect is also shown in Fig. A-1, where the xenon poisoning reactivity is calculated from a standard point kinetic model of xenon poisoning at a constant neutron flux of $2 \times 10^{18} \text{ m}^{-2} \text{s}^{-1}$.

In view of the reactivity balance, the poisoning must be compensated by the reactivity worth of the fine rod:

$$ R_p + R_c = 0 \quad (A-1) $$

For the period of observation on each piece of the data, the worth $R_p$ may be expressed by the product of the differential worth $W$ by the fine-rod displacement $d$:

$$ R_c = W \cdot d \quad (A-2) $$

Inserting Eq. (A-2) into Eq. (A-1), we obtain

$$ W = -R_p/d \quad (A-3) $$

Hence the value of $W$ is determinable by using the measured value of $d$ and the calculated amount of $R_p$.

Figure A-1 shows the results of differential worth $W$ of the fine rod as a function of operation time. At the beginning of full-power operation, a few results indicate relatively high worth of reactivity. This is probably due to a positive reactivity effect ($R_p$) of burnable poison: one gram of B-10 is contained in the two side plates of each fresh fuel assembly. Except for these results, the fine-rod worth is estimated at 55 pcm/cm on the average for 12 pieces of the data. This value is in agreement with the known one of 59 pcm/cm, which has been obtained previously by a reactor period measurement in the following way. Six control members were withdrawn all together by 3 cm, and 800 pcm of reactivity increase was observed, i.e. 267 pcm/cm. The worth of one control member was estimated by dividing 267 pcm/cm by 4.5 instead of 6, because some interference effect of six control members was assumed to be significant, thus being reflected on the effective number 4.5 instead of 6 as it is used by the reactor operators in view of shadowing effect.
DETERMINATION OF DENSITY-NOISE IN A TWO-PHASE-FLOW* BY X-RAY COMPUTER TOMOGRAPHY

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Abstract - A new measuring technique for the observation of two-phase-flow structures is presented. It is based on the principles of x-ray tomography. The density noise signal is measured and characterized by its relevant parameters. The distribution of these parameters are calculated and displayed.

1. INTRODUCTION

In the daily operation of power plants safety is the highest priority. The operators always need to know the immediate working conditions and their tendencies within the components and connecting pipes. Depending on the momentary situation one or two phase flows can be found. The flow pattern, i. e. the distribution of steam in the liquid phase, is mainly responsible for the pressure drop and the heat transfer coefficients.

The often measured average density gives no direct information about the structure of a two phase flow.

The purpose of this work is to evaluate the flow pattern using the local density noise and to display the relevant statistical parameters in the time and frequency domain.

2. TOMOGRAPHY

Tomography is a measuring technique used to find the location of tissue-densities within a body slice. The first theoretical advances were made by Radon (1917), who derived the integral relationships between a distribution and its projections. His analytical work opened the way towards detecting a distribution by measuring its projections.

But no one could materialize this technique and its practical application as long as the fast and high capacity computers were not available.

The medical necessity to non-surgically obtain information about the structure of inner tissue motivated the tomographical research. Extensive measuring efforts were taken in order to produce detailed pictures of stationary layers of tissue using many x-ray sources and hundreds of detectors.

Beside x-ray tomography other physical principles like the NSR-method (nuclear spin resonance) were used.

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The tissue can be understood as being built up by a large number of small volume elements. Each of these elements functions as an absorber with a specific and constant damping of the beam's intensity.

3. TWO-PHASE-FLOW

The non-stationarities which occur within a two-phase-flow can be described by statistical means only. They are reflected in the measuring signal as density noise.

Referring to the model of volume elements, an element acts in this case not as a constant absorber but as a density oscillator. Each detector sees the effects of all the modulation frequencies caused by elements in its beam path.

The question arises whether tomography is a valuable tool to measure and display the pattern distribution within two-phase-flow using the noise signals.

It should be mentioned that a display of statistical parameters is possible if one adheres to the rules for measuring dynamic processes. The calculated image represents the actual flow conditions.

4. MEASUREMENTS

The measuring device used is based on the principles of the medical procedures for tomographic analysis. Contrary to the medical approach the system is smaller yielding a lower resolution of the resulting images; however, this is not relevant for the measurements described here. The device consists of one x-ray source and six scintillation-detectors. It can be rotated 360° around the axis of the test-tube.

The structure of flow determines the lowest frequency of interest and it affects the measuring time and the rotation velocity.

5. ANALYSIS OF THE MEASUREMENT SIGNALS

The tomographic measurement data have to be preprocessed before they can be used to reconstruct the distribution. For this purpose the data are characterized by qualified parameters in the time and frequency domain.

**Time Domain**

1. Moment (Mean Value)  
2. Moment (Variance)  
3. Moment (Skewness)  
4. Moment (Flatness)

**Frequency Domain**

Mean value of the spectral-power density in certain frequency-intervals

These values are taken to be the 'projection-values' of the pattern distribution of the dynamic two-phase-flow.

The following mathematical methods used in medical research to reconstruct density-distributions were examined and later implemented.

- Principle of Summation  
- Principle of Iteration  
- Principle of Convolution  
- Principle of Fourier-Transformation

Each of these methods were tested, including several newly designed modifications, to find the best way to improve the convergence of the calculations and to obtain a reasonable model of the measuring geometry.
It was found that the iterative algorithm worked best in view of the following required criteria.

- Calculation-Stability
- Calculation-Time
- Flexibility in the use of different measure geometries
- Resolution of the resulting images

The calculation time for our best iterative algorithm lies between 20 and 30 seconds. This time can be further reduced.

6. POSSIBLE FORMS OF RESULT PRESENTATION

The results can be displayed in many different ways. The choice depends on the purpose of the analysis. Several examples are displayed.

The measurements included various horizontal and vertical two-phase-flow patterns. The flow variations were adjusted by changing the massflow of the liquid (superficial velocities 0 - 12 m/s) and of the gas (0 - 10 m/s). The experiments covered most of the loop's flow-map.

The detector signals were transformed by fast ratemeters and read on-line into the computer. Afterwards they were processed and stored on a hard disc.

At the end of the measurement the distributions were reconstructed and displayed in different manners.

The following pictures show the developed options for the result presentation using the example of the variance distribution of a measured wavy-flow.

The simple way to display the results is to print the calculated numbers in the form of the numerical distribution matrix used. The image is accurate, but it is time-consuming to see the main information.

The three-dimensional display with a wide-meshed grid gives only limited information, but because of its simplicity it will be used in this presentation.
The use of an isogram offers additional information for a three-dimensional plot.

The three-dimensional display with a close-meshed grid gives the most understandable picture.

The disadvantage of the limited sight is avoided by displaying two more views of rotated (by 90° and 180°) picture. The time needed for its creation can no longer be neglected.
7. EXAMPLES OF TWO-PHASE-FLOW ANALYSIS

The following pictures show noise-tomograms of some typical two-phase-flows.

Two-Phase-Flow

Bubble Flow (Vertical tube)
Waterflow with smaller and larger rising bubbles (1 m/s)

Results in the Time Domain

1) Density  2) Variance  3) Skewness  4) Flatness

1) Higher Density near the tube's wall. The bubbles rise mainly in the tube's center.
2) Clearly distinguishable is the center section with a high number of bubbles.
3/4) Skewness and Flatness describe the asymmetry of a signal-distribution. In this case the distribution-asymmetry appears to be dominant in the area of the tube's wall.

Results in the Frequency-Domain (Spectral-Power-Density)

1) .2 - 2 Hz  2) 2 - 10 Hz  3) 10 - 20 Hz  4) 20 - 100 Hz

1) Slowly rising bubbles, keep mainly to the center-section.
2) Faster incidents of medium size and more equally distributed.
3) Smaller bubbles in the area near the wall.
4) High frequencies concentrate in the area near the wall.
Two-Phase-Flow

Annular-Flashing (Vertical Tube)
Pulsating water ring at the tube's wall and gas with high velocity in the center.

Results in the Time Domain

1) Density  2) Variance  3) Skewness  4) Flatness

1) High density near the wall.
2) Higher flow dynamics near the wall. Flashing water-ring.
3/4) Distribution-asymmetry dominates near the wall.

Results in the Frequency-Domain (Spectral-Power-Density)

1) .2 - 2 Hz  2) 2 - 10 Hz  3) 10 - 20 Hz  4) 20 - 100 Hz

1) Low frequency pulse of the water-ring.
2) Oscillations reach from the wall towards the inner section.
3/4) High frequencies all over the cross-section.
Two-Phase-Flow

Wavy-Flow (Horizontal Tube)
Phase-separation with wavy surface.

Results in the Time Domain

1) Density  2) Variance  3) Skewness  4) Flatness

1) The phase separation becomes obvious.
2) The narrow area of phase fluctuations can be seen.
3) The distribution-asymmetry dominates in the area of the phase boundary.

Results in the Frequency-Domain (Spectral-Power-Density)

1) 0.2 - 2 Hz  2) 2 - 10 Hz  3) 10 - 20 Hz  4) 20 - 100 Hz

1) Dominant wave frequency.
2) Higher frequencies in the way plane.
3) Higher frequencies of lower intensity.
Two-Phase-Flow
Wavy Flow with Foam and sometimes Slugs.
(Horizontal Tube)

Results in the Time Domain

1) Density 2) Variance 3) Skewness 4) Flatness

1) Gradual decrease of density away from the phase boundary.
2) Highest variance caused by the waves. Steep decrease into the water phase, and gradual decrease into the foamy region.
3) Shows the two phases.
4) Shows the area of the waves.

Results in the Frequency-Domain (Spectral-Power-Density)

1) .2 - 2 Hz 2) 2 - 10 Hz 3) 10 - 20 Hz 4) 20 - 100 Hz

1) Distribution of waves.
2) Highest intensity of the waves. Fast wave movement.
3) Incidents of high frequency. The water-ring that sometimes forms in the upper half of the tube.
8. SUMMARY

It has been shown that the tomographic analysis of a stochastic process (e.g., density noise in a two-phase-flow) is possible. The distribution of the relevant statistical parameters can be displayed.

The developed technique is highly suitable for the identification of flow-patterns and it can be used to describe the distribution of the noise components of two-phase-flow.

For this purpose reconstruction algorithms have been implemented and afterwards used to analyse the measured density noise.

This method can be further improved, by
- Reducing the expenses of the measuring device;
- Decreasing the measuring time;
- Decreasing the calculation time.

The application of the described procedure to a geometry of stationary x-ray sources and detectors is possible. This has been tested in additional experiments.

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EARLY DETECTION OF LOCALIZED ONSET OF BOILING

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Abstract - Noise measurement has been often used to detect boiling phenomena in water or other liquids. Although some acceptable correlations have been shown, no in depth study of the relationship between measurement observation and physical boiling process has been satisfactorily presented. This paper shows the results of an experimental investigation on the detection of localized onset of boiling in water on a heated fuel pin. Accelerometers placed on the pin and on the surrounding structures at appreciable distance from the boiling spot were used. A detailed analysis of the results and a comparison with the theoretical bases of boiling has been made in order to extract the actual boiling information from the structure background vibrations. The influence of pressure and temperature have corroborated the proposed relationship between the measured noise and the bubble growing and collapsing processes. The observations conducted on different structures have assessed the independence of the analysis with respect to the structure particular dynamic behaviour. The detection method is shown to be robust and capable to be applied even when accelerometers cannot be placed near the hot zone as in nuclear reactor for instance.

1. INTRODUCTION
Early detection of any perturbation in the normal operation of a process is the aim of reliable instrumentation. Noise analysis has been recognized as a very attractive technique to monitor slight changes, not dangerous in themselves, but potentially responsible for further degradation and damage. This is particularly true in cooling systems where a mismatch between the generated heat and the cooling capability eventually starts in an avalanche way. But cumulative effects, as multiple phase appearance for instance, can quickly lead the process to serious consequences. This paper deals with the detection of boiling onset phenomena. This type of detection is particularly delicate because the boiling occurrence in its onset phase can have a very limited spatial extend and because it does not perturb the process significantly. Only developed nucleate boiling would usually give enough influence on classical instrumentation to be detected, but unfortunately with such a delay that corrective actions become uneffective. Monitoring the boiling noise is an attractive way to allow for sufficiently early detection. Actually, the physical phenomenon that is looked for, starts from the very first appearance of vapour bubbles. The difficulty, however, is that this boiling noise must be observed usually amid a of strong ambient noise level, that eventually impede any reliable detection. The technique presented here is based on the observation of the mechanical noise pattern of the process structure, and the appearance of specific resonances after boiling onset. After a brief review of the boiling mechanism and the boiling noise generation, a series of experimental results are presented. They are compared with theoretical correlations and the influence of coolant parameters and structure geometry is discussed.

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2. BOILING NOISE

Boiling phenomena have been thoroughly studied as one of the most efficient heat transfer mechanisms. Both saturated and subcooled conditions, forced flow and stagnant or free convection have received detailed attention (Cole, 1978). In particular, the mechanisms related to the onset of boiling have generated many theoretical explanations and predictive correlations (Delhaye, 1965 and Butterworth, 1977). It can be defined as the lowest heat flux responsible for the formation of vapour bubbles. On the boiling curve, the onset point limits the single phase convection zone and the local boiling zone. In local boiling, the bubbles are formed on the heated wall and are directly condensed without detachment. A very efficient microconvection occurs in the close vicinity of the heat source, responsible for the higher heat transfer coefficient. This mechanism is essentially apparent in subcooled conditions, which represent most of the commonly encountered situations. It is also to be noted that the onset of boiling does not depend directly on the type of flow, free or forced convection, the whole phenomenon taking place in the wall boundary layer. It is however dependent on the subcooling temperature. Theoretical determination of this heat flux has been proposed, based on graphical constructions on the boiling curve (Bowering, 1962), or on the bubble growth mechanism (Hsu, 1962 and Bergles, 1963). A commonly used correlation has been proposed by Davis and Anderson (1966):

$$\frac{\phi_{onb}}{\rho_v} = -\frac{k_1}{L} \frac{\rho_v}{T_{sat}} \left( T_p - T_{sat} \right)^2$$

(1)

with:
- $k_1$ the thermal conductivity of water [W/cm°C]
- $\phi_{onb}$ the heat flux at the onset of nucleate boiling [W/cm²]
- $T_{sat}$ the saturation temperature [°C]
- $T_p$ the wall temperature [°C]
- $\rho_v$ the vapour density [g/cm³]
- $\sigma$ the surface tension [N/cm]
- $L$ the latent heat of vaporization [Ws/g]

A constant with a value between 1.0 (Davis, 1966) and 1.6 (Hsu, 1962)

A detailed discussion on similar theoretical predictions can be found in (Bouneder, 1987).

Boiling is a process where local rapid volume modification occur in the liquid and the vapour phases. Due to the grow and implosion of bubbles, pressure bursts are produced. They generate acoustic noise both in the liquid and in the surrounding structures. The origin of this noise has been attributed to a series of different mechanisms:

- the pressure pulse during the bubble growth period (Lykov, 1972; Schmidt, 1970 and Robinson, 1974) responsible for a broad band noise around a few kHz.
- the bubble volume oscillations (Strasberg, 1956) with specific frequencies depending on the bubble radius and the liquid parameters (typically around 1 to 5 kHz for water in the pressure range 0.1 to 1 MPA)
- the condensation process with bubble collapses giving wide band frequency pulses up to 10 kHz (Ponter, 1969 and Courbierre, 1981) and even higher for high subcoolings (Berger, 1968).
- the bubble departure rate, that modulates the above mentioned pulses at a much lower frequency, lower than 100 Hz for the 0.1-1.0 MPA range.

The noise intensity is increasing with higher subcooling and decreasing with higher pressure (Ponter, 1969). The experiments refered in the literature are however contradictory about the noise level evolution with the heat flux during the nucleate boiling part of the boiling curve. Depending on the measurement technique, the environmental conditions and the test geometries, proportional relationship (Liska, 1982), leveling off (Tolubinsky, 1974) or progressive diminution (Decrétot, 1986 and Rachedi, 1986) have been reported. The available experimental results are in general limited to the measurement of noise amplitude along relatively large frequency domains. The selection of representative reference levels is difficult and such an approach leads to very delicate test conditions due to the high sensitivity to external perturbations. The tests have also been predominantly performed at atmospheric pressure.

The boiling noise appears in the liquid itself near the nucleation sites but it acts as a source of excitation upon the surrounding mechanical structures. They guide in turn these acoustical waves to the whole environment. The boiling noise can therefore be detected not only by immersed hydrophones, but by measuring the vibration pattern of the surrounding structures. The excitation source has here a wide frequency spectrum and can thus generate a whole set of characteristic vibration modes at specific resonant frequencies. These vibrations are not limited to the immediate vicinity of the boiling place.
3. MEASUREMENT SET-UP

The experimental set-up used for the boiling tests is schematically represented on Fig. 1 (Decréton, 1982). An electrically heated rod is placed in a pressure tube filled with stagnant pressurized water and cooled at the outside by a forced flow of water. The pressure tube is made of aluminium and can withstand a maximum pressure of 2 MPa. The water in the annular gap is demineralized to a fraction of microS/cm and is pressurized by means of a nitrogen upper gas plenum. The heater rod has a stainless steel cladding. Materials in contact with the water are thoroughly cleaned and degreased.

![Diagram of experimental set-up](image)

**Fig. 1  Experimental set-up**

The heating occurs on a part only of the heater height (300 mm), leaving above and beneath this zone long unheated parts. This avoids the end effects on the heat transfer characteristics.
The instrumentation consists in:
- thermocouples placed in the water annulus at different distances from the heating walls
- thermocouples placed at the outside of the pressure tube
- measurement of the pressure of the nitrogen upper plenum
- accelerometers placed on the heater rod at approximately one meter from the boiling zone, on the envelope tube at boiling level and on the supporting structure.
- measurement of the elongation of the heater
- measurement of the characteristics of the outside water cooling.

The measured signals are processed by a computer acquisition system including an accurate Fourier Analyzer. Analog tape recording is possible with a frequency band of 40 kHz.

4. MEASUREMENT RESULTS

Measurements were performed at different pressures between 0.1 and 0.8 MPa. For each of these measurements, the following parameters were recorded:
- the heat flux on the heated wall of the rod (W/cm²)
  Its value is deduced from the total power provided to the heater divided by the heated surface. Axial losses at both extremities have been shown to be low and affect only the last five millimeters. A maximum error of a few percent on the flux is to be considered.
- the heated wall surface temperature (°C)
The wall thermocouples readings have been compared with theoretical predictions of the boiling superheat (MacGregor, 1970 and Collier, 1972) and have shown a very good agreement, of the order of one percent.
- the vibration spectra from the accelerometers
  These signals have been analyzed in the frequency domain, and in particular in the lower range, from 0 to 500 Hz. The spectral power density has been systematically calculated.

Fig. 2 shows a typical result in the frequency domain. It corresponds to the vibrations measured on the supporting structure by an accelerometer sensitive to the radial axis. The distance between boiling spot and measurement is of the order of one meter. The signals are given here for a pressure of 0.3 MPa, at different levels of heat flux ranging from 10 to 70 W/cm². The graphs show the frequency band between 75 and 175 Hz.

Fig. 2 Boiling noise measurements in the frequency domain (N°1, 17.7 W/cm²) (N°2, 31.0 W/cm²) (N°3, 35.4 W/cm²) (N°4, 40.7 W/cm²) (N°5, 45.0 W/cm²) (N°6, 53.0 W/cm²) (N°7, 61.9 W/cm²)
One observes an important background noise component around 100 Hz, generated by the power supply. An added noise component appears around 150 Hz at the onset of boiling, at approximately 30 W/cm². This component increases in amplitude with the heat flux, but decreases in frequency, passing from 150 to 120 Hz when the heat flux increases from 30 to 60 W/cm². The noise peaks are also less acute and present a more damped behaviour.

This observation can be explained by the conjunction of the following interacting phenomena:

- Nucleate boiling generates slight pressure variations, increasing with the heat flux.
- The heated wall temperature increases with the heat flux even in the nucleate boiling region, and leads to a change of the dynamic behaviour of the structure. An increase in the temperature means an increase in the heated rod length and a decrease of its natural frequencies.
- The subcooling decreases with the heat flux, giving larger bubble radii. A decrease in noise frequency is then predicted by the theoretical correlations.
- The bubble density increases with the heat flux, and so does the exciting dynamic energy.

The measured noise spectra are thus clearly excited by the onset of boiling, and form the subharmonic response of the mechanical structure, the boiling acoustical energy being produced in the kHz band, as shown by the theoretical predictions.

Fig. 3 and 4 show the evolution of the boiling peaks as a function of the heat flux and the pressure. This representation presents the same trends mentioned before. The boiling noise frequency decreases with the heat flux and increases with the pressure. However, in both cases, a saturation is observed, with a constant behaviour above 80 W/cm² and 0.3 MPa.

The same observations were already made in previous experiments reported by James (1965).

The following tentative explanation can be given:

- An increasing pressure leads to smaller bubble sizes and therefore larger generated noise frequencies.
- An increasing heat flux leads to higher temperatures and lower subcoolings, and so to smaller natural frequencies of the structure as well as to larger bubble sizes.
- These two phenomena are not linear and act together in reversing direction, giving definite increasing or decreasing slope in parts of the heat flux and pressure range and constant behaviour in others.

![Fig. 3. Influence of the heat flux on the resonant frequency.](image)
The onset of boiling detected by the appearance of these peaks has been compared with alternative predictions. Fig. 5 shows the comparison between the results obtained by noise detection and the prediction by a graphical extrapolation in the boiling curve (considered as the definition of the onset of boiling) and the theoretical model of Eq. (1). This shows the noise detection technique to give very reliable results. The model however fails to predict correctly the results for pressures above 0.4 MPa. This is due to the uncomplete modelization of the microconvection currents appearing for decreasing subcooling at higher pressure (Bouneder, 1987). The validity of the noise detection technique has been further assessed by comparing it with visual estimation in a glass experiment with similar geometry and at atmospheric pressure.

5. CONCLUSION

The technique presented here to detect the onset of nucleate boiling has been shown to be very sensitive even at the beginning of the bubble production. It is based on the monitoring of the mechanical vibrations of the structures surrounding the boiling spot and is focused on the appearance of low frequency peaks in the acceleration frequency spectrum. The detected heat fluxes have been compared with theoretical models and the observed trends have been shown to be accurately predicted. The advantage of such a method is its reliability even when the exact boiling spot is not known. The sensors can actually be placed on the surrounding structures at a certain distance. This renders its application interesting in many industrial processes, including nuclear power plants.

REFERENCES

Fig. 5 Comparison between noise detection and theoretical predictions.


APPLICATION OF A RECURSIVE IDENTIFICATION TECHNIQUE TO NOISE ANALYSIS FOR FUEL PERFORMANCE STUDY

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Abstract - A noise analysis system based on various digital signal processing techniques has been developed for fuel performance studies using test rigs at the R2 reactor in STUDSVIK. This system has been applied to the investigation of the thermal and mechanical behaviour of in-reactor fuel rods with different design parameters and under various fuel irradiation conditions. A recursive identification technique based on an adaptive lattice filter has been developed as a tool for the noise analysis, where the identification model is updated at each data sampling point in order to track time-varying process. The method has been applied to a time constant estimation for the dynamic response of the coolant temperature, i.e., fuel thermal (FT) time constant, and that of cladding elongation, i.e., mechanical response time constant, to a rod power change under both transient and steady conditions during the fuel irradiation. The results demonstrated the usefulness of the technique for studying the fuel behaviour where the thermal and mechanical states are slowly time-varying due to, e.g., fission gas release, creep, swelling and so forth. The paper also presents some results of the FT time constant estimation from various experiments which have been conducted at STUDSVIK.

Keywords - Recursive identification; adaptive lattice filter; fuel thermal time constant; pellet-clad mechanical interaction; fuel performance; noise analysis.

1. INTRODUCTION

Through various applicational studies in the past decade, noise analysis has proved to be a useful tool for investigating in-reactor fuel behaviour and has received considerable interest in the fuel field. The reason for this lies in that, unlike conventional experimental techniques such as step, ramp or scram tests, the noise analysis technique does not require so much perturbation to the experimental environment, thus allowing evaluation of the fuel performance at a given experimental condition. This is very important due to the fact that many of the parameters characterising the fuel performance, e.g., fuel thermal conductivity, heat capacity, gap conductance and so forth are fuel temperature, and thus rod power, dependent.

At STUDSVIK, a noise analysis system based on various digital signal processing techniques has been developed for fuel performance studies using test rigs at the R2 reactor (Oguma, et al. 1986). This system is used routinely to evaluate the so-called fuel thermal (FT) time constant and pellet-clad mechanical interaction (PCMI) of a rod during irradiation. By controlling the coolant temperature, flow, loop pressure, the He-3 absorber pressure, and R2 reactor power, one can achieve quite a wide range of experimental conditions.

A recursive identification technique has been developed and incorporated into the noise analysis system. The method is based on a least squares multivariable lattice filter in which the identification model is updated at each data sampling point using an order and time recursion algorithms (Lee, et al. 1981) in order to adapt it to the time-varying process. The main purpose of applying such
an adaptive method is to obtain better resolution of in-reactor fuel behaviour which may change with time due to complicated mechanisms involved. It is quite beneficial if one can observe the dynamic change of fuel heat transfer characteristics due to fission gas release (FGR) after power increases. Detection of pellet-clad gap increase as a consequence of internal over pressure in a very high burn-up fuel rod is another application. It is the adaptive signal processing technique which is expected to be the most suitable tool to follow up such fuel rod changes. Another advantage in using the recursive method is that one can evaluate the statistical accuracy of the noise analysis result.

The outline of the recursive identification method is given in the next section. Then the general procedure of the FT time constant estimation is presented together with typical results from a number of time constant measurements. Two striking results are presented from application of the recursive identification; one to a power change experiment where the rod power was increased step-wise and then decreased after a hold time at the maximum power, and another to an over power experiment where the rod was subjected to a very rapid and large power increase. In conclusion, some interesting future applications of the technique are presented.

2. RECURSIVE LATTICE FILTER

The recursive identification algorithm we apply is explained briefly. The method is based on the recursive least squares (LS) lattice algorithm developed by Lee, Morf and Friedlander (1981). For an extensive survey on the related subjects, refer to Friedlander (1982).

Given a set of p-variable time series data (TSD), \([X(t); t=1, 2, \ldots, T]\), one can fit a vector autoregressive (AR) model of growing order \(M\) to the data in terms of the covariance lattice form given by

\[
E_M(T) = E_{M-I}(T) - K^f_M(T) E_{M-I}(T-1) \tag{1}
\]

\[
E_M(T) = E_{M-I}(T-1) - K^b_M(T) E_{M-I}(T) \tag{2}
\]

where \(E_M(T)\) and \(E_{M-I}(T)\) are \((T \times M)\)-forward and backward prediction error matrices, \(K^f_M(T)\) and \(K^b_M(T)\) the forward and backward reflection coefficients, respectively. The notation \('\) represents the transpose. The reflection coefficients are adjusted so as to meet the LS criterion for the forward and backward prediction error covariances, resulting in

\[
K^f_M(T) = C_{M-I}(T) R_{M-I}(T)'
\]

\[
K^b_M(T) = E_{M-I}(T)' C_{M-I}(T) \tag{3}
\]

where \(E_M(T)\) and \(R_M(T)\) are forward and backward prediction error covariances and \(C_M(T)\) is the cross correlation of forward and backward prediction errors. The order recursions for the prediction error covariances are obtained as the LS solution as follows,

\[
E_M(T) = E_{M-I}(T) - K^f_M(T) C_{M-I}(T)'
\]

\[
R_M(T) = R_{M-I}(T-1) - C_{M-I}(T) K^b_M(T) \tag{4}
\]

The cross correlation is given from the following time recursion equation (Lee 1980),

\[
C_M(T) = C_{M-I}(T-1) + e_{M-I}(T) g_{M-I}(T-1)/(1 - d_{M-I}(T-1)) \tag{5}
\]

\[
d_M(T) = d_{M-I}(T) + g_{M-I}(T) R_{M-I}(T) g_{M-I}(T)' R_{M-I}(T) \tag{6}
\]
The variable $d_M(T)$ is called the likelihood variable and $1/(1-d_M(T-1))$ in Eq. 4 acts as an adaptive gain, causing the lattice parameters to change quickly. In order for the algorithm to have an adaptive capability for a time-varying process, a forgetting factor is introduced in the algorithm. It acts to apply an exponential data window to the TSD so as to give more weight to recent data points than old ones. Then the time recursion, Eq. 4, is modified to become

$$C_M(T) = \lambda C_M(T-1) + e_{M-1}(T) r_{M-1}(T-1)/(1 - d_{M-1}(T-1)) \quad (4-a)$$

where $\lambda$ is the forgetting factor with a value $0 < \lambda < 1$.

The forward and backward prediction errors at time $T$, i.e., $e_M(T)$ and $r_M(T)$, in Eq. 4-a and 5 are calculated using the lattice prediction filter by applying the sampled data $X(T)$, i.e., $e_0(T) = r_0(T) = X(T)$;

$$e_M(T) = e_{M-1}(T) - K_M^f(T) r_{M-1}(T-1)$$
$$r_M(T) = r_{M-1}(T-1) - K_M^b(T)' e_{M-1}(T) \quad (6)$$

The recursive implementation of Eqs. 2, 3, 4-a, 5 and 6 with initial conditions

$$e_0(T) = r_0(T) = X(T)$$
$$E_0(T) = R_0(T) = \lambda E_0(T-1) + X(T)' X(T) \quad (7)$$

yields the reflection coefficients [ $K_M^f(T)$, $K_M^b(T)$; $M=1, 2, \ldots, Max$ ] as a function of time $T$.

The optimal model order within $M = 1, 2, \ldots, Max$ is determined by the Akaike information criteria (Akaike 1974),

$$AIC(M) = N \log |Q(M)| + 2 N p^2 \quad (8)$$

where $|\cdot|$ denotes the matrix determinant and $N$ the equivalent number of data sample points (Porat 1985) given by

$$N = (1 + \lambda) / (1 - \lambda) \quad (9)$$

$Q(M)$ is the mean of the forward and backward prediction error covariances, i.e.,

$$Q(M) = (E_M(T) + B_M(T))/2 \quad (10)$$

Once the reflection coefficients are obtained, the corresponding AR model at time $T$ is derived by substituting them into the following order recursion equation;

$$A_M(N) = -K_M^f(T)$$
$$B_M(N) = -K_M^b(T)$$
$$A_M(m) = A_{M-1}(m) - K_M^f(T) B_{M-1}(N-m)$$
$$B_M(m) = B_{M-1}(m) - K_M^b(T)' A_{M-1}(N-m) \quad (11)$$

where $m = 1, 2, \ldots, N-1$.

It is noteworthy that the reflection coefficients contain information about all AR models from order 1 to Max. Therefore the AR model of any order up to Max can be deduced from a set of reflection coefficients if it is necessary.

In the computational procedure, the reflection coefficients are calculated using the above mentioned recursion algorithm at each data sample point. Then the AR model is calculated at the end of each data block (consisting of 64 points) and stored in the disk memory to successively evaluate the time behaviour of the time constant.
3. APPLICATION TO FUEL PERFORMANCE STUDIES

3.1. Experimental

Noise experiments have been carried out for fuel performance studies using the on-line data acquisition system at the R2 reactor. Figure 1 shows a schematic representation of the fuel irradiation facility and loop instrumentation together with the noise measurement system. Of various sensors available, coolant thermocouples (Delt-T), a cladding extensometer (EL), and an ex-core neutron detector (N11) are used to measure the coolant temperature variation due to the heat release from the rod, cladding elongation changes and the power produced in the test fuel rod, respectively. During the experiment, these signals were collected on a Winchester disk in the on-line minicomputer (PDP-11/73) after appropriate amplification, lowpass filtering and digitization with an A/D converter. Signals were measured with a sampling rate of 5 or 2 Hz. The purpose of the noise analysis is to estimate the response time constant for the following signal combinations:

- \((N11, \text{Delt-T})\); FT time constant,
- \((N11, \text{EL})\); mechanical response time constant.

In order to determine the FT time constant accurately, it is essential that the input and output signals have a strong enough correlation in the frequency region of interest. Based on previous experience, it was decided to introduce additional pseudo-random-binary-sequence (PRBS) perturbation to the reactivity in order to increase the correlation. For details of the PRBS perturbation system, refer to Oguma et al. (1982).

3.2. Procedure for the time constant estimation

In analyzing the noise data, the TSD are first carefully checked to eliminate, e.g., harmful outliers, trend and so on if they exist. Secondly, the recursive identification is performed for each combination of signals. The AR model is calculated at the end of each data block, that is, every 64 sample points, and stored in a disk file. Thirdly, the step response is calculated by giving a step input to the respective AR model to estimate the response time constant. The time constant is obtained as the time for the step response to reach 63.2% of the final steady value.

3.3. Gamma effect subtraction

The most significant factor influencing the accurate determination of the FT time constant is the gamma heating effect. This is a direct heating of the coolant in the fuel test loop by the gamma radiation generated in the whole reactor core and therefore must be subtracted. As explained in Fig. 2, it is additive to the coolant temperature variation due to the heat release from the rod. This effect can be subtracted in the following way:
- by identifying the gamma heating dynamics independently using empty rod experimental data,
- by inputting the measured neutron signal to the identified gamma heating model, and subtracting the output from the measured coolant temperature to estimate the coolant temperature due to the heat release from the fuel only.

The thus derived gamma-effect-subtracted TSD is used for the FT time constant estimation.

4. RESULTS FROM TIME CONSTANT ESTIMATION

4.1 FT time constant estimation during steady state operation

A considerable number of time constant measurements have been conducted at the R2 reactor for rods with different design parameters and under different irradiation conditions. The time constant measurements for some typical cases are summarized in Table 1 together with parameters characterizing the respective rods.
The rod-3 and -4 are rather unique in that the bore surface of the cladding has facets in order to improve the PCI performance (Wogard et al. 1986). Compared with the standard rod (Rod-1), it appears that this tends to increase the time constant. The rod power increase from 25.4 to 57.2 kW/m caused a significant increase in the time constant (Rod-3). The influence of the design gap is also seen between the Rod-3 and -4. That is, the larger the gap, the bigger the time constant becomes. Comparison of Rod-1 and -5 shows that the rod diameter is an important factor which determines the time constant. To a first approximation, the time constant is proportional to the square of the pellet diameter. The time constant of Rod-1 and its pellet diameter, 10.5mm, could give a value 8.7 sec for Rod-5. This is in relatively good agreement with the experimental result. Rod-2 is filled with Xe and Ar gases which have bad thermal conductivity. The short time constant of this rod appears to contradict expectations. However, a code calculation suggests that the time constant may become shorter in a region where dynamic fuel thermal expansion plays a significant role in determining the gap conductance (Malén 1986).

A typical example of the FT time constant measurement for an 8x8 standard BWR fuel rod (Rod-1) is shown in Fig. 3 according to the noise analysis procedure explained in the previous section. In Fig. 3-b the time constants of the Delt-T response before and after the gamma effect subtraction are compared. There is a clear difference of about 1.5 sec between the two, demonstrating the significance of subtracting the gamma effect. Figure 3-c is the step response of the cladding elongation induced by thermal expansion of the fuel. The estimated FT and elongation time constants by the recursive identification are shown in Fig. 3-d. After the initial transient period of model adaptation, the time constant remains stable and represents only small random variations around the mean value which are mainly due to additional disturbances in the measurements. From the stable part of the curve, the FT time constant is read to be 4.6±0.4 sec.

### 4.2. FT time constant behaviour during a step-wise power variation test

The test fuel rod (Rod-1) included in Table 1 was subjected to a step-wise power variation according to the operation schedule shown in Fig. 4. The FT time constant estimated at each power level is plotted in Fig. 5 together with the mechanical response time constant measured by the cladding elongation signal. The recursive identification was performed for the measurement during the holding period at 44 kW/m linear heat rate (LHR). The result, as shown in Fig. 6, indicates that the time constant of both temperature and cladding elongation increases as a function of time. A sudden jump in the time constant curve at about 80 min is caused by a disturbance superimposed on the data. It is seen by a closer examination of the figure that the rate of increase of the time constant slows down with time. This rod had been irradiated at a relatively low LHR, (at most 32 kW/m) and reached a burn-up of about 20Wd/kgU. Having the past irradiation history of this rod in mind, it is well expected that the FGR should take place after the rod power increase up to such a high level as 44kW/m. Accordingly, the time behaviour of the time constant can be interpreted as the result of the dynamic behaviour of the FGR causing the change in the fuel heat transfer characteristics. It should be noted that the time constant became slightly higher than before when the rod power was decreased down to 30kW/m. This also supports the idea that the FGR in the rod has reduced the gap conductance.
4.3 Cladding elongation response of rods after over power ramp test

The cladding elongation is driven by the axial thermal expansion of the fuel pellet stack, transmitted to the cladding by mechanical interaction or chemical bonding at the burn-up and power levels investigated in this study. Due to the steep temperature gradient across the fuel radius the centre of the pellet can be expected to have the greatest value of its thermal expansion, and the colder periphery the smallest expansion. However, depending on the design and the operating conditions, the point of contact between adjacent pellets controlling the transmission of the axial expansion of individual pellets is not always at the pellet centre. Depending on the mechanical state of the pellet, such as circumferential cracks or high temperature creep, the axial expansion transmitted to the cladding is probably controlled by the thermal expansion of a characteristic radial region of the fuel. It is information about this characteristic radius, as well as the efficiency of the transmission of the fuel strain to the cladding, which we can obtain by noise analysis of the cladding elongation signal. In the case of flat-ended pellets, the characteristic radius for axial interaction would be expected to move outwards from the pellet centre when the fuel temperature is high enough for relatively fast fuel creep. For dished pellets the characteristic radius would be expected to stay the same, or possibly move in slightly due to fuel swelling, assuming that at low initial powers it was at the outside of the dish.

The time constant of the cladding elongation response may vary due to shifting of the characteristic radial position and/or the change of parameters such as the thermal conductivity and heat capacity of the fuel, and the gap conductance.

The over power ramp test is a well established fuel testing procedure in the R2 reactor, where the rod is subjected, after a preconditioning irradiation, to a fast power increase up to a predetermined level, after which the power is kept constant to observe the fuel performance. Noise recordings were made for some rods during the steady power after the ramp operation.

The recursive identification was applied to those experimental data after careful trend subtraction to evaluate the time constant behaviour of the cladding elongation response. Two interesting results are shown in Fig. 8: one exhibiting a stable time constant and the other an increase with time. As seen in Fig. 7, the cladding elongation signal represents very nonstationary behaviour (indicating length reduction with time) probably due to compressive creep of the fuel or sliding between the fuel and cladding. However, the elongation time constant remains stable (Fig. 8-a). This suggests the length reduction may be attributed to the sliding between the fuel and cladding. As explained in the above, the time constant may change owing to changes in the thermal parameters that are caused by, e.g. the FGR and swelling. Increase of the time constant seen in Fig. 8-b may account for such processes progressing inside the fuel.

5. CONCLUDING REMARKS

The present noise analysis for the fuel performance studies indicates that the recursive identification technique applied here has a potential for tracking the dynamic behaviour of the fuel heat transfer characteristics which change in time during in-pile testing. This technique will probably be a suitable tool to investigate the dynamic process of the FP gas release. On the ground of our experiences, it would be interesting to perform long enough measurement to follow the whole process of the FP gas release for different experimental conditions. It is also worthwhile to try the recursive identification to the cladding elongation signal to evaluate dynamic change of the PCTMI state under various types of slow ramp tests.

The FT time constant may be an important parameter in connection with the BWR stability problem. There is a certain concern about reduced stability margin for rods with less diameter than current 8x8 rods, e.g., 9x9 or 10x10 rods because of the shorter time constant. The time constant measurement for such rods should serve to make more accurate evaluation of the stability condition.

Concerning the FT time constant measurements, it is, of course, useful to cover a wide parameter range on the fuel design and irradiation conditions. For example, the time constant measurement for rods with much smaller diameter as
well as for high burn-up rods are the interesting area to be covered. It is believed to be quite profitable if one can establish a time constant data base through systematic accumulation of the data in terms of various parameters such as rod fabrication and irradiation conditions, and post irradiation tests, etc. This would contribute to the fuel performance evaluation.

6. REFERENCES

Oguma R. et al. (1986). Fuel Rod Thermal Performance Studies Based on Noise Analysis, STUDEVSK/NTF(P)-86/08.

Fig. 1 Schematic diagram showing instrumentation used for the noise experiment together with the data acquisition and analysis system.
Fig. 2  Block diagram showing process dynamics associated with the fuel thermal time constant estimation, in which the gamma heating effect is additive and thermocouple dynamics are cascade to the fuel rod heat transfer dynamics.

a) recorded time series data.

b) step response of Delt-T before (upper) and after (lower) gamma effect subtraction.

c) step response of cladding elongation (EL).
d) time constant for Delt-T (left) and for EL (right) estimated by the recursive identification with a forgetting factor $\lambda = 0.998$.

Fig. 3 An example of the time constant measurement for a standard BWR fuel rod.

Fig. 4 Power history in the step-wise rod power change experiment.

Fig. 5 The time constant for the coolant temperature (*) and for the cladding elongation (o) measured at different power levels during the step-wise power change experiment.

Fig. 6 The result from the recursive identification showing increase of the time constant with time during the power holding at 44kW/m; (forgetting factor $\lambda = 0.998$).
Fig. 7 Recorded neutron (N11) and cladding elongation (EL) signals after the over power ramp experiment.

a) stable time constant.

b) time constant increasing with time.

Fig. 8 Time constant estimation by the recursive identification for over power experiments; (forgetting factor \( \lambda = 0.998 \)).
INCORE NEUTRONIC FLUCTUATION MEASUREMENTS IN A LARGE GRAPHITE MODERATED POWER REACTOR

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Abstract - Neutronic fluctuation measurements were made in various locations in the N Reactor core. Although the reactor is very quiet, some interesting features were observed:

1) autospectra that increase in amplitude at low frequencies and exhibit a relatively broad resonant peak at about 0.3 Hz,
2) coherence between all measured locations that shows a relatively broad peak with a value of about 0.8 at 0.3 Hz,
3) zero phase shifts between detectors placed at a common position along the flow stream,
4) linear phase shifts at low frequencies for detectors placed in an upstream/downstream configuration whether or not the detectors are in the same radial location and regardless of the radial locations of the detectors, and
5) a space and time dependent resonant feature at about 1.4 Hz which has strong coherence between selected detectors.

Computer calculations were made to explain some of these observations with only limited success. This paper discusses the results achieved up to this time.

1. INTRODUCTION

An array of neutron detectors was recently installed in the N Reactor to investigate detector responses in preparation for the installation of a much more comprehensive core monitoring system. In addition to continuous static flux measurements over a period of a year, the detectors were used for dynamic measurements and noise measurements.

N Reactor is a 4000 MW thermal, graphite moderated, pressurized-water cooled reactor operated by the Westinghouse Hanford Company for the United States Department of Energy. It serves a dual purpose: producing nuclear products for the Department of Energy and generating up to 860 MW electrical power for use in the Pacific Northwest of the United States. The reactor core is a graphite cubic approximately 10 meters square and 12 meters long. A total of 1003 horizontal Zircalooy-2 pressure tubes containing the cylindrical fuel elements penetrate the graphite stack. Perpendicular to the pressure tubes are 84 horizontal control rods used for normal control and rapid shutdown of the reactor and 640 horizontal Zircalooy-2 tubes used for graphite cooling. Perpendicular to both the pressure tubes and control rods are 107 vertical channels that can be filled with boron carbide balls as a backup safety system for the control rods during a reactor scram.

Just before the N Reactor's scheduled shutdown in January 1987 for an upgrade of the safety systems, an extensive set of noise measurements was made using a preliminary configuration of detectors. Since January 1987, a comprehensive core detector array measuring neutron flux at 360 locations in the core has been installed. The measurements discussed in this paper were made with the preliminary configuration.
2. INCORE MEASUREMENTS

Figure 1 shows the spatial orientation of the detector array. This preliminary configuration included nineteen fission chambers of two different sizes (0.64 cm and 0.48 cm OD and 2.54 cm long) from three different manufacturers and six self-powered neutron detectors (three prompt and three delayed) with diameters of 0.14 cm OD and 40 cm in length.

![Three-dimensional View of N Reactor](image)

Detectors S1-S5 used during single rod scram
Detectors C, D & F used in noise measurements
• Self Powered Neutron Detectors
△ Neutron Fission Chambers

Figure 1. Configuration of Incore Detectors in N Reactor

The detectors have been used to follow reactor transients as well as to make noise measurements. For example, Figure 2 shows the measured response of detectors at positions labeled S1-S5 for the transient initiated by a single rod scram near the reactor front face as shown in Figure 1. A comparison of TWIGL calculations with the observed detector signal is also shown in Figure 2. It is clear from this transient that the reactor responds to local perturbations in a highly space dependent manner. The success of TWIGL calculations in modeling this behavior encourages the use of this code to model other space dependent observations.

![Comparison of Measured Incore Detector Signals with TWIGL Calculations for Single Rod Scram in N Reactor](image)
From the point of view of reactor noise analysis, N Reactor is very quiet with the neutron fluctuations having peak-to-average amplitudes of less than 1% of the mean level and restricted to low frequencies below 2.0 Hz. Detectors throughout the core have coherences with maximums of about 0.8 at 0.3 Hz with the distances between detectors being as large as seven meters.

Figures 3 through 6 show the phase and coherence between pairs of detectors in the reactor. Five measurements are superimposed on each plot. Each measurement is the result of 4 hours of data acquisition. The measurements are very repeatable.

Figure 3 shows zero degrees of phase shift up to approximately 0.8 Hz for detectors placed at a common axial position along the flow stream but separated radially from each other (detectors C and D in Figure 1).

![Cross Spectrum Phases Between Radially Displaced Detectors C and D.](image)

Between 1.2 and 1.5 Hz, some of the phase measurements in Figure 3 become quite erratic. Figure 4 shows the erratic phase is the result of nearly zero coherence between the detectors for some of the measurements. Thus, it appears that a temporal phenomenon is occasionally present that leads to the coherence peaks at about 1.4 Hz.

![Coherences Between Radially Displaced Detectors C and D.](image)
Detector pairs axially displaced (an upstream/downstream configuration) exhibit a phase shift in the cross power spectral density that varies linearly with frequency up to about 0.8 Hz. This is illustrated in Figure 5 for detectors C and F. This phase behavior has been observed whether the detectors are in the same relative radial position or not. For example, detectors C and F are radially as well as axially displaced. This suggests that the phase shift along the direction of the coolant flow is not due to local flow density fluctuations propagating but is related to a more global, space dependent reactivity effect.

Above 0.8 Hz, Figure 5 shows the phase estimates for the axially displaced cases become less consistent with a trend towards a zero phase shift in the frequency range between 1.2 and 1.5 Hz.

![Cross Spectrum Phases Between Axially Displaced Detectors C and F.](image)

The coherence, seen in Figure 6, shows a large, highly reproducible peak with a maximum value of 0.4 at just over 1.4 Hz and less reproducible peaks at 1.3 Hz and 1.5 Hz. While all the measurements between detectors C and F show the peak in the coherence at 1.4 Hz, only some of the measurements between C and G show this same feature. Unfortunately, no measurements were made simultaneously between the pairs (C,D) and (C,F).

![Coherences Between Axially Displaced Detectors C and F.](image)
3. INTERPRETATION

The main questions for interpretation of the data just presented are:

1) an explanation of the main, broad-band coherence that appears in all measurements at frequencies below 0.8 Hz with a maximum at about 0.3 Hz,

2) an explanation of the phase behavior below 0.8 Hz for both the radially displaced detectors at a common axial location and the axially displaced detectors at any radial position, and

3) an explanation of the special features in the range from 1.2 to 1.5 Hz that are both time and space dependent.

Although noise spectra with thermal hydraulic feedback have not been modeled for N Reactor, it is believed that the main, broad-band, essentially space independent, autospectra and coherences below 0.8 Hz are due to the global thermal response of the reactor. This feature is similar to observations made for large PWRs.

The phase behavior is of special interest and a significant effort was made to model this effect. The phase is clearly not due to the propagation of isolated coolant fluctuations because the velocities implied by the linear phase are too high and also because of the high coherence across the core which seems to imply a plane-wave like perturbation propagating from the inlet to the exit.

With the success in modeling the detector response to a single rod scram shown in Figure 2, an attempt was made to calculate the phase shift based on the TWIGL code. To do this, TWIGL was set up to calculate the response of an array of detectors to an inlet temperature perturbation. The perturbation could be introduced in either a single coolant channel or a group of channels including up to all inlet channels. The computer results show that even single channel perturbations results in a near plane wave response as it travels down the core. Also, the speed of the reactivity-induced perturbation is significantly higher than the coolant flow.

Figures 7 and 8 shows a result from a TWIGL calculation for an inlet perturbation to all flow channels. The time traces in Figure 7 are the calculated detector responses at the same locations as detectors C and F whose measurements are shown in Figures 5 and 6.

![Figure 7. TWIGL Calculated Time Response for Detectors at Locations C and F.](image)

These time responses were Fourier transformed and used to calculate auto and cross spectra. These results are shown in Figure 8. Of interest is the calculated phase shift which is not linear and appears to be much larger than observed. However, the calculated phase up to 0.5 Hz is about \(-20^\circ\) which is in substantial agreement with measurements. Above 0.5 Hz, the calculated phase continues to increase to a value of about \(-50^\circ\) at 1 Hz and \(-90^\circ\) at 1.5 Hz. In contrast, the measured phase shift reaches a nearly constant value of about \(-20^\circ\) in the range from 0.5 to 1.2 Hz and then tends toward a 0° phase shift above 1.2 Hz.
While the calculations represent fairly well the actual physical behavior of N Reactor in the region of maximum observed coherence below 0.8 Hz, for the higher frequencies where smaller coherences are observed, there is a substantial discrepancy between calculations and measurements. These results lead us to conclude that the physical explanation for the observed phases at very low frequencies may be modeled by existing codes but the higher frequency behavior is caused by a physical phenomenon that is not being modeled by the calculation.

Figure 8. Cross Spectrum Phase Between Detectors C and F as Calculated by TWIGL

The features seen in the measurements at 1.4 Hz are not modeled in the computer calculations and may be responsible for lack of agreement in phase behavior seen between 0.8 and 1.5 Hz. The intermittent nature of this feature as observed between one pair of detectors (Figure 4) contrasts with the regular nature observed from another detector pair (Figure 5). The question remains as to the true spatial and temporal characteristics of the 1.4 Hz feature. The investigation of the cause of this feature will be undertaken when the reactor restarts with a full incore detector array.

4. INTERIM RESULTS

N Reactor exhibits space dependent reactor noise features. The observed phase shifts are only partially explained by a two dimensional dynamics calculation. A special resonant feature at about 1.4 Hz has been observed.

The reactor has been in a shutdown since January 1987. A full array of incore detectors has been installed during the shutdown period. After restart of the reactor, this array of incore detectors will be used to fully investigate the interesting features that preliminary measurements have revealed.
EXPERIMENTAL INVESTIGATIONS OF CORRELATIONS BETWEEN NEUTRON POWER AND FUEL ELEMENT OUTLET TEMPERATURES OF SUPER PHENIX 1

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ABSTRACT - The commissioning procedure of the 3000 MWth SUPER PHENIX 1 reactor provided for the first time the opportunity to extensively study neutron power and outlet temperature noise phenomena at a large fast breeder reactor. The noise measurements cover power levels between 15 and 100 % nominal power. Two ex-vessel and one in-vessel neutron chamber were used for the neutron noise measurements. The temperature noise was measured using standard Cr-Al thermocouples as well as intrinsic SS-Na thermocouples at a few SA outlets.

The measurements have shown that at SPX1 there is very little correlation between neutron power and outlet temperature noise. The reasons for this have been identified by a detailed analysis of noise data. It is concluded that the major part of the outlet temperature noise is not related to the heat generation process in the SA's. It is presumably produced by turbulent coolant flow in the upper plenum.

As a consequence, heat transfer parameters of the fuel elements could not be obtained from noise measurements. They had to be determined by the analysis of the outlet temperature response to fast transients of reactor power (scram).

The predominant temperature noise component generated outside of the fuel elements will make it extremely difficult to detect local loss of cooling in a SA or developing fuel failure at an early stage by monitoring outlet temperature noise characteristics. However, the results also indicate that the possibilities for early failure detection by outlet temperature noise analysis could be improved significantly by placing the TC's closer to the SA outlets.

The response time of Cr-Al thermocouples could be derived from power spectral densities of Cr-Al and SS-Na thermocouple noise.

KEYWORDS
Fast reactor; SUPER PHENIX 1; core surveillance; fuel element performance monitoring; LMFR neutron noise; outlet temperature noise; heat transfer coefficient; thermocouple response time; intrinsic thermocouples.

INTRODUCTION
The power-induced component of a subassembly ('SA') outlet temperature signal can be derived from a simple lumped-parameter fuel element model /1/. At steady state reactor operation, the linear frequency spectra of measured outlet temperature noise $\Delta T(w)$ and neutron power noise $P(w)$ are related by the SA power-to-outlet temperature transfer function $H(w)$ and the thermocouple (TC) transmission characteristics $G(w)$:

$$\Delta T(w) = G(w).H(w).P(w)/(a.F)$$ (1)
To a good approximation / 1 / the transfer functions can be written as:

\[ H(\omega) = \frac{1 - \gamma}{(1+j\omega \tau_f).\left(1+j\omega \tau_c\right) - \gamma} \cdot \exp{-j\omega \tau} \quad (2) \]

\[ G(\omega) = \frac{1}{1+j\omega \tau_c} \quad (3) \]

wherein:

\[ \gamma = \frac{k}{k+2sF} \quad \text{feedback parameter} \]

\[ \tau_f = \frac{C_f}{k} \quad \text{fuel time constant} \]

\[ \tau_c = \frac{\gamma C_c}{k} = \frac{C_c}{k+2sF} \quad \text{coolant time constant} \]

\[ \tau_t = \text{coolant transit time core midplane - SA outlet} \]

\[ \tau_r = \text{thermocouple response time} \]

\[ \omega^2 = 1 \]

with the SA-parameters:

\[ F = \text{coolant mass flow rate} \ [\text{kg/s}] \]

\[ C_f, C_c = \text{heat capacity of fuel, coolant in a SA} \ [\text{J/K}] \]

\[ s = \text{specific heat of coolant} \ [\text{J kg}^{-1} \text{K}^{-1}] \]

\[ k = \text{average fuel-to-coolant heat transfer coefficient} \ [\text{W/K}] \]

The fuel element transfer function \( H(\omega) \) is represented by a second-order low-pass characteristics with feedback. The exponential term in Eqn. 2 accounts for the transit time of the coolant through the SA. It will be dropped in this context because it is a pure phase factor and only the gain of transfer functions and power spectral densities will be considered.

The coolant time constant of a typical fast breeder SA is less than the fuel time constant by more than a factor of ten. Thus, the transfer function (2) can be approximated by a first-order low-pass characteristics (3) with the modified fuel time constant:

\[ \tau = \frac{\tau_f}{1 - \gamma} = \tau_f + \frac{C_f}{2sF} \]

Consequently, measuring the SA transfer function and fitting to it the model transfer function provides a time constant which is the sum of the fuel time constant and a small correction term which in general can be neglected. For SPX1 SA's at nominal operating conditions it is less than 10% of the fuel time constant.

In order to facilitate the intercomparison of neutron power and outlet temperature noise of SPX1 measured at different reactor operating conditions as well as with earlier measurements at ENK II and PHENIX only power spectral density ('PSD') curves normalized to signal or temperature mean values (SS-Na) will be presented ('NSPD'). Inlet temperature PSD's will be normalized to core temperature rise.

From step response measurements / 2 / it is known that fuel time constants are in the order of 5 s and TC response times less than 1 s even for Cr-Al TC's. Therefore, we can also neglect the TC transfer function \( G(\omega) \) in Eqn. 1. Then we obtain for the mean values of SA power \( P(0) \) and temperature rise \( \Delta T(0) \):

\[ \Delta T(0) = P(0)/(s\cdot F) \]
Thus, the normalized linear spectra are given by:

$$ A_n(\omega) = H(\omega) \cdot P_n(\omega) \quad (4) $$

If power fluctuations were the only source of outlet temperature noise normalized PSD's of neutron power and outlet temperature noise would be identical below the cut-off frequency $f_c = 1/(2\pi) f = 0.03$ Hz. This was true for KKN II and PHENIX. However, at SPX1 this temperature noise component is largely dominated by other noise phenomena as will be seen later.

Equation 1 can also be used for calculating dynamic reference temperatures to be compared on-line to measured outlet temperatures for sensitive monitoring of individual fast breeder fuel element performance. This technique has been successfully tested earlier at the fast sodium-cooled test reactor KKN II and the French breeder prototype PHENIX / 3 / . The aim of this work was to check whether this method would equally apply to commercial-size fast breeder reactors.

**SPX1 NEUTRON AND TEMPERATURE INSTRUMENTATION**

For the measurement of neutron power several ionisation chambers inside and outside the reactor vessel are available. The three in-vessel chambers are mounted one upon another on a rig which can be moved vertically inside of the central SA. The used chamber was about 2 m above the core midplane during the measurements reported here. The ex-vessel chambers are located 9.5 m below core midplane at angular distances of 120° from each other. They are connected to the fission process in the core by neutron guide tubes installed at blanket element positions at the core edge i.e. 12 SA rows from the central core axis. Evidently, ex-vessel chambers are less efficient than those inside the vessel. Furthermore, in view of the large core and the distance between core and detector positions space dependent and local effects might be encountered in measured neutron power noise.

All of the 376 fuel elements (including 18 dummies) and the 72 breeder SA's of the first blanket row as well as the 21 control rods are equipped with a triple temperature probe which consists of two conventional Cr-Al thermocouples ('CTC') and one intrinsic SS-Na TC ('ITC'). These probes are located about 10 cm downstream of the SA outlets. This makes them susceptible to temperature fluctuations which might be produced by turbulent cross flow and mixing of sodium at different temperatures in the upper plenum.

The CTC's forming part of the reactor safety system are normally not accessible to direct recording. However, during a limited period of time the signals of eight of them were made available for direct recording together with inlet temperature (CTC) and neutron power signals. This was in a series of experiments performed for investigating higher than expected temperature fluctuations at some blanket and peripheral fuel SA outlets. For this reason four of the chosen CTC's were on blanket positions, three of them on peripheral fuel elements and only one was measuring the outlet temperature of a fuel SA near to the core centre. Noise measurements with this fixed set of CTC's could be performed up to 50 % of nominal reactor power only. Intrinsic CTC's were not available during this period because their signal channels had to be used for the CTC's.

The noise measurements from 50 % to full power have been made with the ITC's only. However, mean and RMS values of CTC readings could be calculated from the digital records of all of the outlet temperatures available through the plant's core surveillance system CORA. The intrinsic CTC's have a much shorter response time than the CTC's (1 s). Therefore, they also measure the high-frequency temperature noise which is not transmitted by the CTC's due to their low-pass transmission characteristics. For the measurement of the SA power-to-outlet temperature transfer function this is of no use because the fuel time constants to be determined are even larger than the response time of the CTC's. But if there is sufficient high-frequency content in the temperature fluctuations the response time of the CTC's can be determined by crosscorrelating ITC with CTC noise.

The cold reference electrode of the ITC's is located 1 m above the SA outlets. The sodium temperature is not kept constant. It rather follows with some delay the mean core outlet temperature variations.

As a consequence, the temperature fluctuations which are induced simultaneously at all SA outlets by fluctuations of reactor power will be partly or even completely eliminated from the ITC signals depending on frequency and SA temperature rise. On the other hand, any local temperature variation at the location of the reference electrode would be found in all the outlet temperature signals. Therefore, good correlation between neutron power and ITC temperature signals cannot be expected. On the contrary, cold reference temperature fluctuations would increase the coherence between outlet temperatures itself even for non-neighbouring SA's.
Although the ITC reference electrode is also mounted in a triple probe together with two CTC's the signals of the CTC's were not yet available for the measurements. They will be connected later to the SPXI noise analysis system ANABEL to enable the elimination of cold reference fluctuation effects on ITC noise.

DATA ACQUISITION AND ANALYSIS SYSTEMS

The simultaneous readings of all Cr-Al thermocouples can be recorded by the SPXI core surveillance system CORA / 4 / in the form of digital records on magnetic tape. The maximum sampling rate of outlet temperatures is 1 Hz. Simultaneous sampling of neutron power is not possible. Therefore, the CORA temperatures cannot be crosscorrelated with neutron power. Only statistical parameters and power spectral densities of inlet and outlet temperatures can be derived from CORA data files.

The only system which provides analog signals of neutron chambers and thermocouples is the on-line noise analysis system ANABEL / 5 / . Only eight of the 469 ITC's at a time can be connected to ANABEL by a remote-controlled scanner. Since there is no multichannel spectrum analyser included in the ANABEL system the experimental SA surveillance system KASUMOS / 6 / developed at KFK was used for the measurements and the analysis of both ANABEL signals and CORA data files.

The signals are preamplified and low-pass filtered before being transmitted from the reactor building to ANABEL in the SPXI computer room. There they are connected in parallel to ANABEL and to the KASUMOS preprocessing inputs. In the signal preprocessing 3 neutron and 9 temperature signals (1 inlet temperature) are multiplexed, preconditioned and digitised before being recorded on magnetic tape for later off-line analysis with the KASUMOS main computer at the Cadarache nuclear research centre.

The measurements with the CTC's were made in the same way when they were available. The signal sampling rate was 19 Hz in all cases.

NOISE MEASUREMENTS

From the various noise analyses performed at power levels between 15 and 100 % nominal power only a few examples will be given in order to illustrate typical SPXI noise phenomena.

Neutron noise

In this work neutron noise was of interest as a driving force of SA outlet temperature noise only. The power noise spectral densities measured at various reactor operating conditions are compared to the corresponding outlet temperature noise PSD's. The characteristics of SPXI power noise were investigated in more detail by Le Guillou et al. / 7 / .

In Fig. 1 the normalized APSD of an in-vessel neutron chamber is shown for 23, 50 and 100 % n.p. and primary pump speeds of 200, 400 and 425 r/m, respectively. Thus, in the measurements at 23 and 50 % n.p. coolant temperatures were nearly the same whereas primary coolant flow and reactor power differed by a factor of two. For the 50 and 100 % n.p. NPSD's the coolant flow rate was about the same but coolant temperature rise and power differed by a factor of two this time. It is seen that at constant (low) coolant temperature the power noise increases significantly with coolant velocity. It is reduced when the core temperature is increased by rising the reactor power at constant primary flow.

Resonance peaks at higher frequencies probably due to mechanical vibration phenomena obviously depend on coolant velocity rather than on core temperature. The peak at $8.10^{-4}$ Hz is due to inlet temperature fluctuations caused by feedwater control oscillations which exist at reduced power only.

The outlet temperature noise decreased strongly with increasing coolant flow at constant coolant temperature as can be seen from the NPSD's shown in Fig. 2. This might indicate that there is less of turbulent mixing of coolant between SA outlets and TC positions at higher coolant velocity. The zone of mixing then might be shifted further downstream by the more stiff SA outlet jets. One could conclude from this that the temperature noise produced outside the SA's could be eliminated from the signals by approaching the TC's closer to the SA outlets.

In Figs. 3 and 4 NPSD's of the ex- and in-vessel chambers are intercompared for 50 and 100 % power and coolant flow. Except for one ex-vessel chamber the NPSD's do not change significantly between 50 and 100 % n.p. However, at frequencies between 0.01 and 0.1 Hz the PSD of the in-vessel chamber is always higher than the ex-vessel PSD's. The latter reach their detection (plus signal channel) noise level at about 1 Hz. The detection noise level of the in-vessel chamber is nearly a factor of ten lower which corresponds to its higher efficiency.
Some of the peaks at higher frequencies are common to all neutron chambers, others exist only in ex- or in-vessel chambers. For some of them there is a frequency shift with changing reactor operating conditions. The normalised cross power spectral densities ('NCPSD') between in- and ex-vessel chambers agree fairly well. As it is shown in Figs. 5 and 6 the coherence between the ex-vessel chambers is much better than between in- and ex-vessel chambers. This indicates that higher in-vessel noise might not be due to real power fluctuations.

**Outlet temperature noise**

Only some major aspects of SPF1 SA outlet temperature noise are discussed here as far as they are of interest for fuel element performance monitoring. A systematic study on outlet temperature noise characteristics of the whole SPF1 core was done by Girard et al. /8/.

Some examples of temperature noise NAPSD's measured at 50 % and full power using CTC's as well as ITC's are shown in Figs. 7 and 8. The temperature signals were calibrated using a sensitivity of 15 μV/K and 41 μV/K for ITC's and CTC's, respectively. It is evident that ITC and CTC noise are completely different from each other even though the experimental conditions were not exactly the same in both cases. The intrinsic TC's noise is higher than that of the CTC's over the whole frequency range.

This discrepancy results from ITC and CTC properties. First, the CTC's clearly show a roll-off at $f > 0.2$ Hz due to their finite response time which is much smaller for the almost prompt-responding ITC's. Therefore CTC's do not transmit high-frequency temperature fluctuations. The higher ITC noise level for $f < 0.1$ Hz is due to fluctuations of the reference temperature as will be seen later. So, for $f < 0.1$ Hz only the CTC's measure real temperature noise whereas beyond this frequency ITC signals are valid.

However, the most important phenomenon observed is the large discrepancy between a calculated estimate of power-induced outlet temperature noise and the actually measured one. This is demonstrated in Fig. 9 wherein calculated and measured NAPSD's of outlet temperature noise are shown together with power noise NPSD's. The outlet temperature NAPSD resulting from power fluctuations was derived from the ex-vessel chamber's NCPSD by multiplying it by the square modulus of the SA transfer function (Cf. Equ. 4) using a fuel time constant of 5 s /2/.

Obviously, the power noise is only a minor driving force of SA outlet temperature noise. The predominant outlet temperature noise is most likely generated by turbulent cross flow and mixing of Na in the upper plenum. This assumption is supported by the fact that outlet temperature noise was decreasing with increasing coolant velocity as discussed before (Fig. 2).

This interpretation seems to be further confirmed by the fact that at the core-blanket boundary up to ten times higher RMS values than the average were found. The larger difference in coolant velocity and mean outlet temperature between fuel and blanket elements is particularly favouring the generation of temperature noise by cross flow and mixing of sodium.

For the peripheral fuel SA's it could be shown explicitly that the measured temperature noise is not related to the heat generation process in them. Figure 10 shows the outlet temperatures of one SA at the core edge and one near to the core centre which have been measured during a reactor scram from 50 % n.p. It is seen that the temperature fluctuations of the edge SA did not disappear when the coolant temperature rise decreased to zero as at the central SA outlet. They rather seem to decrease with the decreasing difference between core and blanket outlet temperatures.

Outlet temperature noise could not be measured simultaneously with CTC's and ITC's for the reasons mentioned before. However, measurements at 50 % reactor power could be performed successively with both types of thermocouples. In Fig. 11 the NAPSD's of outlet temperature noise from these measurements are shown together with a calculated PSD of CTC temperature noise. The calculated PSD was obtained from the ITC noise PSD by multiplying it by the square modulus of the TC transmission characteristics (3). The theoretical curve was fitted to the measured one by adjusting the response time in the transfer function (3). The best match was obtained with a TC response time of 150 ms.

From the good agreement of these curves for $f > 0.2$ Hz it is concluded that ITC signals represent real temperature noise beyond this frequency.

Below their cut-off frequency CTC's measure the temperature fluctuations correctly so that the ITC noise must be faulty. This was proven by measuring the outlet temperature response to small power steps of about 10 % rise time. Chromel-Alumel TC's gave correct temperature variations whereas intrinsic TC signals did not change with reactor power.
Consequently, there is no or very little correlation between neutron power and ITC noise at low frequencies. Since there are no power-induced temperature fluctuations beyond 0.2 Hz because the fuel time constant is in the order of 5 s ITC and neutron noise signals are completely uncorrelated. On the other hand, good correlation was found between different ITC's even when they were separated by several rows of SA's. It is concluded therefore that the additional noise component in the ITC signals is due to temperature fluctuations at the reference point.

The wide frequency range of coolant temperature noise in the upper plenum offers the possibility to measure the transfer function of the CTC's. Classical crosscorrelation techniques could not be used with intrinsic and Cr-Al thermocouples because their signals were not simultaneously available. However, an estimate of CTC response times could be obtained by matching measured and calculated PSD's of CTC noise as discussed before.

In Fig. 12 the APSD's of four CTC signals are normalised at about 0.1 Hz. Up to this frequency the TC transfer function should be flat. The different shape of the curves at higher frequencies is determined by the individual TC's response time. Minimum and maximum values of it have been estimated by matching the theoretical curve to the limiting PSD curves in Fig. 12. In this way it was found that the four available CTC's on fuel element outlets have a response time between 150 and 450 ms. These values are significantly less than the response time specified for the TC fabrication. This means that the chromel-alumel thermocouples have a better than expected thermal contact to the coolant. Smaller than specified response times were also found by the step response measurements for most of the outlet thermocouples.

CONCLUSION

The major part of temperature noise measured above SFXI SA outlets is not related to fluctuations of reactor or SA power. Measured temperature noise is presumably due to turbulent cross flow of sodium in the upper plenum. The results also indicate that this large background noise could be reduced by placing the thermocouples closer to the SA outlets.

Temperature noise measured with intrinsic thermocouples is valid for frequencies higher than 0.1 Hz only. At decreasing frequency there is an increasing contribution from cold reference temperature fluctuations. These can be eliminated in future experiments by separately measuring the reference temperature noise and subtracting it from the outlet temperature signals.

As a consequence of the lack of coherence between neutron power and SA outlet temperature noise thermal hydraulics fuel parameters as fuel time constant, fuel-to-coolant heat transfer coefficient or fuel-clad gap conductance could not be obtained by the analysis of reactor noise at steady state reactor operation. Fairly large and fast power variations were necessary for this purpose.

The very small ratio of power-induced temperature noise to that generated outside of the SA's will make it extremely difficult or even impossible to detect local loss of coolant in a SA or developing fuel failure at an early stage by monitoring outlet temperature noise characteristics.

For the same reason the sensitivity of conventional SA outlet temperature monitoring (mean value) relative to local loss of cooling in a SA cannot be increased by comparing measured outlet temperature to model predicted individual dynamic reference temperatures as proposed earlier. However, for transient reactor operation this might still be useful.

The presence of high-frequency coolant temperature noise and the availability of intrinsic thermocouples offers the possibility of measuring the response time of the Cr-Al thermocouples in SFXI.
Fig. 1 Change of SPX-1 neutron power NPSD with coolant flow and temperature (in-vessel chamber)

Fig. 2 Change of outlet temperature NPSD with coolant flow and temperature (central SA)

Fig. 3 SPX-1 power noise PSD's of in- and ex-vessel neutron chambers; 50% n.p.

Fig. 4 SPX-1 power noise PSD's of in- and ex-vessel detectors; 100% n.p.
Fig. 5 Coherence between in- and ex-vessel neutron chambers; 50% n.p.

Fig. 6 Coherence between in- and ex-vessel neutron chambers; 100% n.p.

Fig. 7 Outlet temperature noise APSD's from Cr-Al thermocouples; 50% n.p.

Fig. 8 Outlet temperature noise APSD from intrinsic thermocouples; 100% n.p.
Neutron power and outlet temperature correlation

Fig. 9 Outlet temperature noise as measured with CTC's and calculated from power noise

Fig. 10 Coolant temperatures during reactor scram

Fig. 11 PSD of temperature noise measured with CTC and calculated from ITC PSD

Fig. 12 Adjustment of CTC response times to measured APSD's
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DETECTION OF COOLANT TEMPERATURE NOISE IN SPX1 USING INTRINSIC HIGH FREQUENCY THERMOCOUPLES

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Abstract - Under the European agreement on Fast Reactors, France, the Federal Republic of Germany and the United Kingdom have been collaborating on the measurement, analysis and modelling of high frequency temperature noise signals from intrinsic stainless steel/sodium thermocouples on the Superphenix reactor during its commissioning. A description is given of the experimental programme, and of the analysis using the k-value method. This is followed by a summary of initial modelling using the STATEN computer code to predict the temperature noise levels at the thermocouple.

1. INTRODUCTION

For a number of years it has been proposed (Bentley, 1974; Firth, 1977) that internal blockages in a fast reactor subassembly can be detected at an early stage by monitoring broad band temperature noise levels at or near the subassembly outlet, even though the mixed mean temperature rise due to the blockage may be negligible. Experimental programmes to assess the viability of the technique have been carried out in a number of countries (Weinkoetz, Krebs and Martin, 1982a; Girard and Buravand, 1982; Firth and Conroy, 1986) and supported by theoretical modelling (Krebs and Bremhorst, 1983; Firth, 1977; Hughes, Overton and Wey, 1986). However, the use of temperature noise as a diagnostic technique has not been widely accepted to date because of the lack of long term reactor measurements using sensors with a sufficiently high bandwidth.

The commissioning of the Superphenix reactor, with its comprehensive array of intrinsic thermocouples and associated data acquisition system, ANABEL, has provided an opportunity to extend the assessment of temperature noise, both from an experimental and a theoretical point of view. Under the European collaboration on Fast Reactors, the UK and FRG have been collaborating with CEA France to obtain and analyse noise data, and to carry out theoretical modelling of the experimental programme.

This Paper reports on the temperature noise experimental programme on Superphenix which was carried out during late 1986 at power levels of between 50\% and 100\% of full power. A synopsis of the experimental results is presented, along with some details of analysis using the KfK k-value method and modelling work using the STATEN computer code.

2. EXPERIMENTAL PROGRAMME AND DATA COLLECTION

The temperature of the sodium coolant as it leaves the core of the 1200 MW(e) fast breeder reactor Superphenix is monitored by a comprehensive array of temperature probes. These probes are mounted 100mm above the mouth of each of the fuel sub-assemblies (see Fig. 1) and of the
first, innermost, ring of breeders giving a total of 448 instrumented sub-assemblies. Each temperature probe contains two conventional chromel-alumel thermocouples and one stainless steel electrode which forms one arm of an intrinsic thermocouple.

![Diagram of SPX1 above-subassembly geometry](image)

**Fig 1. SPX1 above-subassembly geometry**

The intrinsic stainless steel/sodium thermocouple consists of two wire electrodes immersed in liquid sodium and measures the thermoelectric emf due to the difference in temperature between the two bi-metal junctions. If one of these junctions is in a region of sodium at a known constant reference temperature, then the thermocouple will measure the temperature of the other junction relative to this reference. Since the temperature to be measured is that of the sodium at the junction itself, the thermal time delay involved is very small and a frequency response of greater than 100Hz can be achieved. In a practical system however, it is necessary to low pass filter the output signal at less than 50Hz to eliminate mains interference, but the final bandwidth remains very much greater than that of any other thermocouple type, making the intrinsic device suited to the measurement of temperature noise. In SPX1 the intrinsic thermocouple signals are low pass filtered to give a bandwidth of 10Hz while the 3mm chromel-alumel thermocouples have a bandwidth of 0.2Hz. The two chromel-alumel thermocouples in each probe are connected to the two independent temperature monitoring systems which are able to shut down the reactor in the event of a large excursion in the outlet temperature of one or more sub-assemblies in the core. The intrinsic thermocouples are connected via a remote controlled selector unit which selects eight signals from 448, and these are then preamplified and fed to the on-line noise analysis system ANABEL (ANALyse des Bruit En Ligne). After further filtering and amplification the signals are output to an analogue magnetic tape recorder. ANABEL controls the selection of thermocouples, provides the signal processing parameters and details of reactor operating conditions, and monitors the basic statistical functions of the waveforms (RMS, skewness, kurtosis).

In order to confirm that the signals seen by the intrinsic thermocouples were in fact genuine temperature noise, a comparison was made with the adjacent chromel-alumel thermocouple signals for a number of sub-assembly positions at 50% reactor power. To make the comparison it was necessary to low pass filter the intrinsic signals at 0.2Hz and adjust the amplitude scale in...
order to obtain the optimum match with the chromel-alumel signal. The corresponding waveforms are shown in Figure 2 for breeder sub-assembly 2842 over a time period of 62 seconds. It can be seen that the two signals are very similar, indicating that both types of thermocouple are seeing the same temperature fluctuation field.

Previous work with stainless steel/sodium thermocouples has shown that the determination of sensitivity through the measurement of mean voltages is complicated by the presence of steady differences of electric potential in the circuit. The spurious voltages do not affect the measurement of temperature noise for which a valid sensitivity can be determined. The sensitivity derived from this matching was 16μV/°C which is close to the value expected at a temperature of 500°C. Figure 3 shows the intrinsic thermocouple signal before and after filtering. It can be seen that there is considerable high frequency activity superimposed on the fluctuations seen by the chromel-alumel thermocouple and this appears to be random in amplitude and frequency as expected of a genuine temperature noise signal.

Sample recordings of the temperature noise from the intrinsic thermocouples were made during the run up to full power of SPX1 between August and December 1986. These recordings were made during:

(1) Steady state power levels (15%, 60%, 80%, 90%, 100%)

(2) Power raising (50%-80%, 80%-100%)

(3) Non-uniform movements of the reactor control rods

Of particular interest among this last group of conditions were a series of tests which involved raising one control rod while lowering all the others together in order to maintain constant overall power. The resulting hot region of core gave rise to steep temperature gradients across the nearby sub-assemblies and provided an "abnormal" condition similar to that envisaged due to a flow restriction within the sub-assembly.
As an example of the data collected, Figure 4 shows a map of RMS temperature noise levels for all sub-assemblies at a steady state power of 80%. Noise levels are generally between 0.5 and 1.0°C RMS across most of the fuel but rise sharply at the fuel-breeder boundary where the maximum levels of almost 4°C were measured.

Under the European fast reactor collaboration agreement, this data was made available to KfK Karlsruhe Germany and CEGB Berkeley UK for further analysis and comparison with the predictions of the model codes.

3. EVALUATION OF THE K-VALUE

Temperature fluctuations are generated by radial temperature gradients within fuel subassemblies. The nature of the fluctuating signals is stochastic which means that they can be described by statistical signal analysis.

One of the significant statistical parameters is the r.m.s. value of temperature fluctuations. Numerous out-of-pile experiments (Weinkoetz, Martin and Krebs, 1979; Weinkoetz, Krebs and Martin, 1982a; Krebs and Brehmhorst, 1983) and in-pile investigations on KNK II (Weinkoetz, Krebs and Martin, 1982b) were performed with respect to the r.m.s. value. The essential outcome of these investigations was the discovery of a linear relationship between the r.m.s. value (σ) of temperature fluctuations and the coolant temperature rise (ΔT). This led to the definition of a characteristic bundle coefficient, hereafter termed k-value, as the ratio between the r.m.s. value of low pass filtered temperature fluctuations and the coolant temperature rise along the subassembly.

\[
k\text{-value} = \frac{\text{low pass filtered r.m.s. value (σ)}}{\text{coolant temperature rise (ΔT)}}
\]
The r.m.s. value depends on the reactor operation conditions (reactor power, mass flow) and radial temperature distribution at the bundle outlet which is also dependent on the geometry of the fuel subassembly. By contrast the k-value indicates only a change of geometry such as a flow blockage, or else a change of power tilt by control rod displacements.

Based on these considerations, the measured fluctuating temperature signals at the fuel subassembly outlet of some SPX1 subassemblies have been analysed.

Figure 5 shows the pump speed V and the coolant temperature rise $\Delta T$ as a function of the reactor power between 50% and 100%. The mass flow thus increases linearly with reactor power up to 72% at constant coolant temperature rise. Above this power level the coolant temperature rise increases proportionally to the reactor power while the mass flow remains constant.

Figure 6 shows the r.m.s. values of three subassemblies located along a radial row of the SPX1 core. At constant mass flow, the r.m.s. values of two subassemblies increase with the coolant temperature rise $\Delta T$, while the r.m.s. value for subassembly 2931 is nearly constant. At constant coolant temperature rise and increasing mass flow respectively, the r.m.s. values are constant except for subassembly 2231. Similar results have been obtained analysing other subassemblies of the radial row mentioned above (numbers 1931-2931).
Fig 5. SPX1 power/coolant temperature rise control curve

Fig 6. Variation of r.m.s. temperature with reactor power for a number of SPX1 subassemblies

It can be seen that the r.m.s. values are not strictly proportional to the coolant temperature rise. Therefore, the calculated k-values are not quite constant between 50% and 100% of full
power level (Fig. 7). According to the definition, the k-value should be independent of reactor power. To find out the reason for this behaviour different influence parameters have been analysed. These investigations allow the following statements to be made:

- There was no cross-correlation between temperature signals from neighbouring subassemblies.
- It has been proved that the reference point of the intrinsic thermocouples has no influence on the measured r.m.s. level of fluctuating temperature signals from the subassemblies.
- The requirement of the definition of the k-value is given by 10Hz low pass filtered temperature signals.
- The subassembly outlet temperature does not influence essentially the e.m.f. sensitivity of the intrinsic thermocouples.
- The ratio of measured r.m.s. values of temperature noise is small compared to KNK II and out-of-pile measurements.

![Graph showing variation of k-values with reactor power for a number of SPX1 subassemblies](image)

Fig 7. Variation of k-values with reactor power for a number of SPX1 subassemblies

Related to these results, it may be assumed that a significant part of the measured r.m.s. value is not correlated with the coolant temperature rise. One possibility (Edelmann, Girard and Massier, 1987) is the influence of cross-flows (for certain areas of the core) above the subassembly outlet.

In addition to the evaluation of three selected temperature signals, the k-values are presented in Fig. 8 for the subassembly numbers 1931 to 2931 which are located along a radial row of the SPX1 core. The k-values are compared for the power levels 60% and 80%. Within this power range the mass flow and the coolant temperature rise change. Nevertheless, only marginal variations of the k-values occur. The k-values near the wall of the core are much higher owing to the strong power tilt in this region. However, the coolant temperature rise of all subassemblies increases uniformly with the reactor power as shown in Fig. 9.
Fig 8. K-values for subassemblies along a radial row of the SPX1 core

Fig 9. Coolant temperature rises for subassemblies along a radial row of the SPX1 core
4. THEORETICAL MODELLING

4.1 The STATEN code

The STATEN computer code (Overton, Wey and Hughes, 1982), developed by CEBG, uses a Monte Carlo method to model turbulent flow in the open wrapper of a Fast Reactor subassembly between the pin bundle exit and the plane of the thermocouple. The degree of turbulent mixing is controlled by the cross-stream turbulence intensity specified to the code which is used to allocate instantaneous cross-stream velocities to marker particles released in coplanar batches into the flow. These velocities are mutually weighted both spatially and temporally to achieve the desired turbulence length scales and spectral form of the velocity distribution. The particles carry representative temperatures downstream from an initial specified temperature profile, and explicit heat transfer calculations are performed at each of a large number of discrete steps between the initial plane and the final (thermocouple) plane, some distance beyond the subassembly outlet. A simulated temperature signal of full bandwidth can thus be built up at any specified point in the flow, and in particular at the thermocouple location. To facilitate comparison with measured temperatures from limited bandwidth measurements, the signals can be digitally low-pass filtered before computation of parameters such as r.m.s. noise levels and the skewness and kurtosis of the temperature distribution. For axially non-uniform geometries, such as the Superphenix subassembly open wrapper, STATEN is used in linked axial stages, which permits axial variation in turbulence conditions caused by changes in the geometry (grid, venturi etc) to be accommodated.

During its development, STATEN has been tested against out-of-pile measurements in sodium and in water, at CEBG (Wey, Overton and Hughes, 1982), UKAEA and KfK (Dubuisson et al, 1987). Comparisons in sodium have been most successful when a detailed distribution of turbulence intensity was available.

4.2 Application to a Superphenix subassembly

For the purpose of modelling the region of interest of the Superphenix subassembly, five linked axial stages were used (Fig. 10) with no heat flux modelled across the wall of the subassembly. Turbulence intensities based on previous modelling experience in a convergent and divergent geometry were used; these are presented as a ratio to the bulk velocity in Table 1, along with geometry and velocity details. As an initial study, two cases were considered: namely Subassembly reference 3230 at 80% power with the nearby control rod 1 at each of its two extreme positions during the experiment (z=228mm and z=0mm).

For these cases, CEA supplied detailed temperature profile calculations for the pin bundle exit, one example being shown in Fig. 11. The near-symmetry of the profile meant that the 2D version of the STATEN code could be used without much loss of accuracy. A sample length of 2048 x 1ms was generated by the code for the thermocouple location, and as stated above, was low-pass filtered (10Hz, first order) before analysis for mean and r.m.s. temperatures, and for skewness and kurtosis. This ensured that the assessed frequency response of the measurement system was correctly incorporated in the model. In order to ascertain the spatial distribution of these parameters, simulated measurement positions were also placed across a diameter at the end of each of the five linked STATEN stages, and data were obtained at each position.

4.3 Results of the STATEN simulation

The results for the two simulated cases are given in Table 2 together with the analysis of the corresponding experimental results. Since the two cases do not differ greatly, the results are presented graphically mainly for one case (control rod position z=228mm).

Figure 12 shows the calculated initial mean temperature rise profile at the pin bundle exit (cf Fig. 11), and the prediction of mean temperature at the thermocouple plane.
Fig 10. SPX1 subassembly geometry modelled by the STATEN code

Fig 11. Estimated pin bundle subchannel exit mean temperatures

SUBASSEMBLY 323D: Control Rod Z=228mm

Fig 12. Predicted mean temperature development
Figures 13, 14 and 15 show, respectively, the r.m.s. temperature, skewness and kurtosis predictions at the sensor plane, derived both from the 10Hz filtered signal and from the

![Graph showing r.m.s. temperature](image)

**Fig 13.** Predicted r.m.s. temperatures at the thermocouple plane

![Graph showing skewness](image)

**Fig 14.** Predicted temperature skewness at the thermocouple plane
unfiltered signal. The analysed values from Superphenix measurements are also indicated, on the assumption that the thermocouple is located on the axis of the subassembly. Figure 16 illustrates the predicted development of filtered r.m.s. temperature for the centre
line of the subassembly, as well as for a line 16mm each side of the centre line in the 2D geometry. In addition, Fig. 17 compares the centre line axial development of filtered r.m.s. levels, and the Superphenix measurements, for each of the two control rod positions.

4.4 Discussion

The mean temperature profiles of Fig. 12 demonstrate the small degree of decay at the centre line (1.4°C) occurring during the transit time of 0.15s. This decay has, however, generated about 0.5°C r.m.s. (filtered) noise at the centre line (Fig. 13). Significantly more is generated at the positions of maximum mean temperature gradient (up to 5°C r.m.s. filtered). The corresponding centre line value derived from Superphenix analysis is 0.75°C r.m.s. (filtered). From Fig. 17 it can be seen that for the two different cases considered, the different levels of r.m.s. are echoed by the STATEN predictions. It should be noted, however, that there is a tolerance of several millimetres on the radial positioning of the thermocouple, and it can be seen that the Superphenix measurement is within the predictions at, say, +/- 10mm. The Superphenix skewness value of -0.44 is nearer to zero than that predicted by STATEN, but kurtosis is closely predicted despite the erratic radial profile of this fourth order statistic. The lack of smoothness of the predicted skewness and kurtosis profiles may in fact be attributed to the limited length of the simulated signal. On the axis of the subassembly, the r.m.s. temperatures can be seen to increase from the assumed zero level at the pin bundle outlet to the thermocouple (Fig. 16), whereas off the axis, an initial increase is followed by a decrease where the geometry expands and dissipation of noise exceeds its production.

The variation in radial thermocouple location may be seen as a possible explanation of the variations in values of r.m.s., skewness and kurtosis across the core seen in Fig. 4 (except for the large and systematic change across the core/breeder boundary).

As well as thermocouple positioning, there are other factors which could be responsible for the small differences between the predicted and Superphenix values. These include any significant deviation of individual thermocouples from the assumed uniform sensitivity, and any errors in the estimates used in STATEN of the turbulence intensity and initial temperature at the pin bundle exit. In particular, any variation from the flatness of the profile near the centre line would tend to give a predicted skewness nearer to zero at the thermocouple and thus nearer to the Superphenix value. At the same time it would also increase the predicted r.m.s. level to nearer that measured.
Bearing all these points in mind, it may be said that, in the limited number of cases considered so far, STATEN provides a reassuringly close prediction of the measurements made on the reactor. However, it must be remembered that a great deal more verification will be necessary for subassemblies at different core positions and powers in order to ensure that the modelling is sufficient, and to assess whether unmodelled aspects such as cross-flow above the subassembly outlet are important. The predicted radial profiles of mean and r.m.s. values at the thermocouple plane could have important implications and are worthy of future experimental verification.

Table 1. Geometry and flow details used in the STATEN simulation

<table>
<thead>
<tr>
<th>STAGE LENGTH (mm)</th>
<th>INITIAL DIAMETER (mm)</th>
<th>FINAL DIAMETER (mm)</th>
<th>AV BULK VELOCITY (m/s)</th>
<th>TURBULENCE INTENSITY RATIO</th>
<th>RADIAL MACROSCALE (mm)</th>
</tr>
</thead>
<tbody>
<tr>
<td>1</td>
<td>263</td>
<td>172</td>
<td>70</td>
<td>9.86</td>
<td>0.10</td>
</tr>
<tr>
<td>2</td>
<td>568</td>
<td>70</td>
<td>70</td>
<td>12.68</td>
<td>0.05</td>
</tr>
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<td>3</td>
<td>168</td>
<td>70</td>
<td>92</td>
<td>9.64</td>
<td>0.04</td>
</tr>
<tr>
<td>4</td>
<td>75</td>
<td>92</td>
<td>92</td>
<td>7.34</td>
<td>0.04</td>
</tr>
<tr>
<td>5</td>
<td>255</td>
<td>92</td>
<td>125</td>
<td>5.40</td>
<td>0.04</td>
</tr>
</tbody>
</table>

Table 2. STATEN predictions and Superphenix analysis of the temperature distribution at the thermocouple for subassembly 3230 at 80% power

CONTROL ROD POSITION z=228mm

<table>
<thead>
<tr>
<th>MEAN TEMP RISE FROM INLET (DEG C)</th>
<th>RMS TEMP (DEG C)</th>
<th>SKEWNESS</th>
<th>KURTOSIS</th>
</tr>
</thead>
<tbody>
<tr>
<td>PIN BUNDLE EXIT</td>
<td>173.0</td>
<td></td>
<td></td>
</tr>
<tr>
<td>UNFILTERED PREDICTED SIGNAL</td>
<td>171.63</td>
<td>1.13</td>
<td>-1.78</td>
</tr>
<tr>
<td>FILTERED (10Hz) PREDICTED SIGNAL</td>
<td>171.63</td>
<td>0.56</td>
<td>-0.82</td>
</tr>
<tr>
<td>SUPERPHENIX ANALYSIS</td>
<td>-</td>
<td>0.75</td>
<td>-0.44</td>
</tr>
</tbody>
</table>

CONTROL ROD POSITION z=0mm

<table>
<thead>
<tr>
<th>MEAN TEMP RISE FROM INLET (DEG C)</th>
<th>RMS TEMP (DEG C)</th>
<th>SKEWNESS</th>
<th>KURTOSIS</th>
</tr>
</thead>
<tbody>
<tr>
<td>PIN BUNDLE EXIT</td>
<td>154.90</td>
<td></td>
<td></td>
</tr>
<tr>
<td>UNFILTERED PREDICTED SIGNAL</td>
<td>153.81</td>
<td>0.78</td>
<td>-1.49</td>
</tr>
<tr>
<td>FILTERED (10Hz) PREDICTED SIGNAL</td>
<td>153.81</td>
<td>0.39</td>
<td>-1.11</td>
</tr>
<tr>
<td>SUPERPHENIX ANALYSIS</td>
<td>-</td>
<td>0.67</td>
<td>-0.59</td>
</tr>
</tbody>
</table>
5. CONCLUSIONS

This paper has presented a brief summary of collaborative European work on temperature noise, centred around an experimental programme mounted on Superphenix. During the commissioning phase on this reactor, good quality temperature noise signals, derived from intrinsic stainless steel/sodium thermocouples, were obtained. The signals, from 448 instrumented subassemblies, have been analysed for r.m.s. levels and other statistical parameters, and show a sensitive response to the many different levels of reactor power and control rod positions over which recordings were made.

The availability of temperature noise data has provided an opportunity for assessment of the k-value (normalised r.m.s. temperature) which has been shown to remain close to an expected constant level as reactor power and mass flow vary.

In addition, an initial study has attempted to predict the temperature noise levels at the thermocouple for one subassembly under two different conditions using the STATEN simulation code. Despite using estimates for the turbulence levels in the subassembly, the Superphenix values were closely predicted. However, a great deal more verification work remains to be done before an algorithm can be proposed to detect pin bundle blockages by monitoring temperature noise levels.

6. REFERENCES


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NOISE ANALYSIS FOR THERMAL STUDIES OF LMFBR FUEL RODS

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ABSTRACT - This paper presents the results of the noise analysis of two fuel rod irradiation experiments - NILOC I and NILOC II - with LMFBR mixed (U,Pu) nitride fuel, irradiated in the High Flux Reactor (HFR) at Petten. The fluctuations of the thermocouple signals in the shroud together with reactor neutron detector signals have been analysed in time and frequency domain. Bi-variate auto-regression modelling results in experimental step response times. Corresponding temperature responses and step response parameters have also been calculated using a computational code for a two-dimensional (R-Z) time dependent finite difference thermal model, for a perturbation with a step in heat generation. Comparison between the experimental and calculated results gave satisfactory information for validation of the model and the experiment. The potential of the application of noise analysis methods in fuel irradiation experiments is discussed.

KEYWORDS
Temperature Noise; Fuel rod performance; HFR Reactor; Irradiation Experiments; Reactor Noise.

1. INTRODUCTION

The LMFBR fuel rod performance during an operational transient has been studied by the Commission of the European Communities Joint Research Center -JRCP- at Petten. At present a wide variety of irradiation experiments are being conducted (Moss and Zeisser, 1986 and Plitz et al, 1986) to study fuel rods of high density, under long irradiation periods and under transients of different rates. Most of the fuel rods are mixed oxide, carbide or nitride fuels in highly instrumented capsules.

In the framework of LMFBR fuel studies two irradiation experiments (NILOC I and NILOC II) have been conducted in the High Flux Reactor (HFR) of JRCP during HFR cycles 86.11 and 87.02 with three slightly different mixed (U,Pu) nitride fuel rods in each experiment.

The main interest of the experiment was to determine the temperature profile during the entire irradiation period of 25.5 days under the high level of irradiation and to compare measured temperatures with model calculations. Also temperature noise measurements have been carried out for these experiments to gain insight in the thermal characteristics of the fuel rod under stationary irradiation. NILOC I and NILOC II signals from the axial thermocouples located in the molybdenum shroud were measured together with reactor safety channel neutron detector signals, using an existing on-line data acquisition system.

The signals (DC and AC parts) were measured continuously during the whole period of irradiation, partly recorded and analysed in time and frequency domain.

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From the experiment, heat transfer characteristics have been determined (Türkcan, Oguma, Hayashi, 1984; Türkcan and De Vries, 1987) using the transfer function from heat source fluctuations to temperature fluctuations in the fuel. In fact the heat source fluctuations are not directly measurable, but they are proportional to the neutron power fluctuations of the reactor. These fluctuations are global in the frequency range of interest and therefore they can be measured by a neutron detector at any place in the reactor.

It can be seen from the calculations of the transfer functions that small changes in the reactor power result in changes in the temperature signals at frequencies below 0.5 Hz. The transfer function has been characterized by the time constant between reactor power and fuel rod temperature. This time constant has been estimated by bi-variate autoregression analysis of the two signals. This has been done for all axial thermocouples in each of the three loops of both NILOC I and NILOC II.

Several numerical calculations have been carried out using a two-dimensional (R-Z) finite difference time dependent thermal heat transfer code. Step response calculations were compared to the measured ones. It will be demonstrated how the noise analysis technique can contribute to a better understanding of the behaviour of fuel rods under operational conditions.

2. EXPERIMENTAL FACILITY AND DATA ACQUISITION

Test section
The NILOC I and NILOC II experiments were carried out at the 1/5 MW thermal, materials testing High Flux Reactor (Röttger et al., 1986) at Petten. NILOC consists of three LMFBR fuel pins placed in the TRIXO carrier in reactor core positions G3 and G5. The basic difference between the fuels are the fuel rod diameters. Fuel is mixed nitride (U,Pu)N screened with a cadmium plate from thermal neutrons of the reactor.

The TRIXO carrier is a reloadable standard facility. Cooling of the channel is provided with reactor primary water. Molybdenum was preferred as the shroud material because of its superior thermal conductivity and dimensional stability. The primary containment is surrounded by a stainless steel secondary containment tube, having a small helium gas gap between the tubes, by which the fuel rod clad temperature may be adjusted.

Fig. 1 shows the reactor TRIXO capsule and data acquisition system. Fig. 2 shows a schematic outline of a fuel rod and the radial position of the thermocouples at each fuel rod. Axial positions of the thermocouples at each fuel rod are shown in fig. 1 and in tables 2 and 3. In the position where the NILOC I and NILOC II have been placed (G3 and G5) the nuclear heating is about 7.1 W g⁻¹.

Data acquisition
The reactor power monitoring signals NC5, NC7, (neutron detectors) together with reactor control rod position signals and thermocouple signals of the experiment have been connected to the on-line data acquisition system (see fig. 1) built for transient overpower experiments TOP (Türkcan and De Vries, 1987). The thermocouple signals were first amplified with a gain of 100, then preconditioned by setting gain and filtering of the AC part of the signals (above 0.01 Hz). The DC (mean value) and the AC signals were sampled with 8 and 1 sample/seconds respectively. To obtain the maximum possible dynamic range in the fluctuating part of the signals, the gain of the amplifiers were adjusted between x128 and x256. Data were simultaneously recorded on digital tape, and analysed during the whole experimental time. The noise data acquisition system has been built for 16 channels simultaneous recording. The NILOC experiments have 14 thermocouples for each fuel rod. Therefore not all the signals could be recorded at once, but a suitable selection of 16 at a time had to be made. The DC part of the signals indicate the conditions of the reactor and the NILOC experiment. The AC part of the signals gives very detailed information about the dynamic behaviour of the experiment. A sophisti-
cated data-acquisition system for signal conditioning and data collection has been used in these experiments, which consist of:

a) analog and digitizing part controlled by a micro-processor in down loading;

b) signal transmission and preprocessing part, by modems and mini-computer PDP11/24

c) data analysing part by digital computer VAX 11-750 with array processor (FPS5100).

Pre-analysis of the data is performed for checking stability by calculating mean and standard deviations of the measured signals. A great deal of the recorded data is analysed in time and frequency domain with parametric and non-parametric methods.

Fast Fourier Transform (FFT) calculations have been carried out to derive fundamental signatures from the measurements, like APSD, CPSD and transfer functions between power and fuel rod temperatures. The final analysis has been made by the Multi Variate-Regression (MAR) modelling technique to estimate the dynamic characteristics of the fuel rod through the evaluation of the system response (Türkcan, Oguma, 1984).

The inherent fluctuations (the noise) are used to identify the system dynamics, by calculating first the model coefficients for input/output relations. In our calculations the reactor power signal is taken as an input and the thermocouple signal of the fuel rod as an output. The step response is calculated by integrating the calculated impulse response calculated for the identified model. The step response time is derived as the time for the step response to reach 63.2% of the final state.

Also another method of analysis has been applied. A second order transfer function has been fitted to the gain and phase of the experimental transfer function, obtained from either the FFT analysis or the identified MAR model, for details see Türkcan and De Vries (1987). The transfer function parameters can also be used to derive dynamic parameters to be compared with heat transfer model calculations, like step responses. This has not yet been done, but results will presumably not differ from MAR step responses. Anyway, the quality of the transfer function fitting will also be demonstrated in this paper.

3. EXPERIMENTAL RESULTS

Figure 3 and figure 4 show DC- and AC-signals of one of the NILOC II experiments. The neutron detector signal is proportional to reactor power (45 MW-theoretical). The mean values of all signals are constant. Total fuel rod power (gamma heating and fissile heating) is determined by flow rate in the cooling channel, the temperature difference in the cooling channel and the heat capacity of the fuel.

Corresponding AC signals are shown in fig. 4, where the noise signals of the neutron detector (NC5) and the noise signals of the TC5 in loop 1 have been displayed. Fig. 4 shows that the neutron detector noise (reactor noise) has much higher frequency content than the thermocouple signal, and that the signals are correlated. This observation is proved by the calculation of APSD and coherence functions. Fig. 5 gives reactor spectra APSD up to 32.0 Hz, while the coherence between the reactor power and temperature is given in the same figure. It is clear that the coherence between reactor power and temperature in the fuel rod does not extend above 1.0 Hz. Therefore further analysis of the experimental data has been carried out up to 2.0 Hz, by filtering the measured signal by a sharp digital filter at 2.0 Hz.
Table 1. Temperature T, the step response time τ, and step response amplitude A of the NILOC I experiments.

<table>
<thead>
<tr>
<th>THERMOCOUPLE POSITION (mm)</th>
<th>LOOP 1</th>
<th>LOOP 2</th>
<th>FUEL TOP</th>
<th>LOOP 3</th>
</tr>
</thead>
<tbody>
<tr>
<td></td>
<td>T(℃)</td>
<td>T(τ)</td>
<td>A</td>
<td>T(℃)</td>
</tr>
<tr>
<td>+143</td>
<td>461</td>
<td>4.43</td>
<td>0.83</td>
<td>467</td>
</tr>
<tr>
<td>1/4</td>
<td>481</td>
<td>4.84</td>
<td>0.80</td>
<td>462</td>
</tr>
<tr>
<td>+64</td>
<td>501</td>
<td>3.17</td>
<td>0.69</td>
<td>503</td>
</tr>
<tr>
<td>3/4</td>
<td></td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>-23</td>
<td>451</td>
<td>4.96</td>
<td>0.87</td>
<td>466</td>
</tr>
<tr>
<td>-54</td>
<td>427</td>
<td>3.79</td>
<td>7.03*</td>
<td>465</td>
</tr>
<tr>
<td>-89</td>
<td>467</td>
<td>3.01</td>
<td>0.65</td>
<td>463</td>
</tr>
<tr>
<td>12</td>
<td>446</td>
<td>3.06</td>
<td>0.75</td>
<td>438</td>
</tr>
</tbody>
</table>

DATE: 22-23 Dec. 1986
TIME: 16:57 - 03:03
FILE: NILOC67

DATE: 23 Dec. 1986
TIME: 10:23 - 16:29
FILE: NILOC68

TIME: 17:34 - 09:05
FILE: NILOC69

*) Probably scaling error.

As an example figure 6 shows the APSD of the various thermocouple signals. The shapes of the APSD functions are the same, but the amplitudes depend on the position of the thermocouple and the generated heat from the fuel. After calculating FFT transforms for all channels for the total length of data, the correlation function (ACF and CCF) were also determined.

Bi-variate auto-regression modelling has been used to estimate the dynamic characteristics of the fuel rod through the evaluation of the step response time. This has been applied to each pair of signal combinations of neutron detector and (axially located) thermocouples. The neutron detector signal is used as an input and the thermocouple signal as an output in the identified model, where the step response is calculated.

Figure 7 gives an example for the model and step response calculation for the pair of signals of NC5 and L1/TC5 of NILOC II. Complete results of the step response times and amplitudes for the NILOC I and NILOC II are given in the tables 2 and 3. These tables contain mean temperatures and step response times derived from bi-variate analysis. Measured standard deviations of the step response times are of the order of 8% for the total measurement time.
The analysis of data by least-squares fitting to transfer function gain and phase shows that the fitting is perfect between 0.02 and 0.64 Hz. Also here no clear change with fuel pellet diameter can be observed.

Experimental results can be compared:
- step response times of the NILOC I are larger than for the NILOC II;
- step response time amplitudes, $A$, are smaller in the NILOC I, indicating that the linear power rates of the fuel is smaller than in NILOC II.

There is no clear change in step response time, in the axial direction due to change in the pellet diameter.

Table 2. Temperature $T$, the step response time $T_s$, and step response amplitude $A$ of the NILOC II experiments.

<table>
<thead>
<tr>
<th>THERMOCOUPLE POSITION (mm)</th>
<th>LOOP 1</th>
<th>LOOP 2</th>
<th>LOOP 3</th>
</tr>
</thead>
<tbody>
<tr>
<td></td>
<td>$T_{(1)}$</td>
<td>$T_{(2)}$</td>
<td>$A$</td>
</tr>
<tr>
<td>$+1.43 / -6$</td>
<td>529</td>
<td>3.92</td>
<td>1.92</td>
</tr>
<tr>
<td>$+0.64 / -6$</td>
<td>537</td>
<td>2.99</td>
<td>1.51</td>
</tr>
<tr>
<td>$0 / -7$</td>
<td>584</td>
<td>2.42</td>
<td>1.79</td>
</tr>
<tr>
<td>$-0.23 / -10$</td>
<td>590</td>
<td>2.27</td>
<td>1.44</td>
</tr>
<tr>
<td>$-0.54 / -11$</td>
<td>573</td>
<td>2.59</td>
<td>1.55</td>
</tr>
<tr>
<td>$-0.89 / -12$</td>
<td>607</td>
<td>3.17</td>
<td>1.58</td>
</tr>
<tr>
<td>$-1.49 / -13$</td>
<td>564</td>
<td>3.66</td>
<td>1.78</td>
</tr>
</tbody>
</table>

DATE: 16-17 FEB 1987
TIME: 17.26 - 09.30
FILE: NILOC88

DATE: 15 FEB 1987
TIME: 11.00 - 22.40
FILE: NILOC86

DATE: 17-18 FEB 1987
TIME: 10.50 - 03.50
FILE: NILOC89

4. CALCULATIONAL MODEL

The two dimensional, time-dependent heat conduction differential equation,

$$ k_{R,Z} \frac{dT_{R,Z}(t)}{dt} + Q_{R,Z}(t) = (\rho c_p) \frac{T_{R,Z}(t)}{t} $$

and suitable boundary conditions, with

$T_{R,Z}(t)$ - time-dependent temperature distribution

$Q_{R,Z}(t)$ - heat generation

$\rho c_p$ - heat capacity

$t$ - time,

are solved by the finite difference method in $R,Z$ geometry.
For the given geometry \((R,Z)\) (see fig. 10) and material properties, including gas specification, steady state temperature distributions have been calculated for NILOC I and NILOC II specifications. The calculation has also been carried out for the different heat generations (linear power rating of the fuel rod), and for different gas mixtures (He/Ne).

After the steady state calculations a certain stepwise increase (1%) in heat generation is introduced and iteration and time integration is carried out until the new steady state temperature distribution has been obtained. In most of our calculations \(n\Delta t = 20\) s with \(\Delta t = 0.1\) s. Step response time is determined from \(\tau_{R,Z}(t)\) when the temperature reaches 63.2% of the final value.

Figure 11 shows radial and axial temperature distributions at the steady state. It is interesting to see that only the \(\tau_{R,Z}\) in the fuel region shows changes of fuel diameter. Figure 12 shows normalized temperature step responses for different axial thermocouple positions.

During the measurements the reactor power was kept constant, but due to the different diameter of the fuels of the NILOC I and II, the generated linear power was different.

Step response time is calculated for all positions of the TC's for various values of linear power. Figure 13 shows calculated results for two thermocouple positions for both fuel rods NILOC I and NILOC II. Results of these calculations show that the step response time decreases with increasing linear power to an asymptotic constant value. This change is due to change in the heat transfer coefficients.

5. CONCLUSIONS

A comparison can now be made between measurements and calculations. The shapes of the step responses as derived from noise data (fig. 7) and from the theoretical model (fig. 12) agree reasonably well. A more quantitative comparison is provided by the step response times where direct comparison is possible, experimental values are included in figure 13, showing a fair agreement. However there seems to be a slight overestimation of the experimental values (of the order of half a second). From other experiences (Oguma et al., 1986) it is known that this might be attributed to the gamma heating of the fuel rods. Methods may be derived to check this experimentally. Nevertheless, the quantitative agreement appears to be sufficient for verification purposes.

To summarize, the experience obtained in the NILOC I and in the NILOC II fuel rod measurements indicates various possibilities where application of noise analysis during the irradiation of the experimental fuel is justified. The data acquisition technique and the developed methods are sufficient for the following purposes:

- Generally speaking the noise characteristics of the signals permit more accurate determination of the experimental conditions, by in-situ testing of measured channels and monitoring the condition of the fuel rods and the reactor.

- The measurement of step response time gives insight into the thermohydraulic behaviour of the fuel rod. In principle they make comparison possible between different types of fuel or different rod dimensions. In the NILOC I and NILOC II experiments it has been demonstrated that the different fuel gives different step response time for the same reactor condition. Step response times measured at the top and at the bottom of the fuel are about the same. Higher mean temperature due to higher fuel rod diameter (in the NILOC II) corresponds with lower step response time. It should be noted that a detailed axial analysis of step response times is probably difficult for pellet type fuels, due to local variations in gap sizes due to pellet eccentricity.
Measured results can be used to verify thermohydraulic calculations for the specific fuels under the irradiation conditions, including linear power rating and He/Ne gas combinations. The first results, presented in this paper, clearly show this. More extensive experiments are desirable for completness, by including fuel thermocouples, thermocouples in the coolant channel in the fuel region, and neutron detectors in the TRIOX carrier. Tests must be included to investigate the gamma heating effect.

ACKNOWLEDGEMENT

The authors express their thanks to the HFR operational staff, to the experimental group of NILOC, especially to Mr. M. Beers for his help during the experiment and the experiment’s sponsors, Trans Uranium Institute, Karlsruhe. Special thanks are due to Mr. W.H.J. Quaadvliet for his assistance in the experiment and analysis of data. The authors also thanks Mr. J.P. Verhoef for his programming effort on the time dependent computational code and the programming of step response time calculations.

REFERENCES


Fig. 1 Data acquisition and processing system together with HFR schematics, with NILOC fuel rods and cross-section of the TROI X carrier.

Fig. 2 Schematic diagram of fuel rod (left) and the radial positions of the thermocouples (right).
Thermal studies of LMFBR fuel rods

Fig. 3 Three neutron detector DC signals (upper) and mean temperature signals of the loop 1 (L1) and loop 2 (L2) (lower).

Fig. 4 Display of the noise signals: (upper) neutron detectors, (lower) temperature noise.

Fig. 5 APSD function of the neutron noise of the HFR and coherence between neutron detector and thermocouple TC5.

Fig. 6 APSD functions of the various thermocouple signals of the NILOC II experiment.
Fig. 7 The calculation of Akaike's Information Criterion (AIC) versus MAR-model order (left) and the calculated step response (right) from neutron detector signal to temperature signal of TC5.

Fig. 8 Combination of NC5 and TC5 with the parameters.

Least squares fitting results of second order transfer function GAIN and PHASE to the signals of reactor power and temperature on the shroud. The experimental transfer functions from (NC5-TC5) $A_i, B_j, t_i$ are fitted to model parameters.
Fig. 10 The NILOC model indicating materials and thermocouple positions.

Fig. 11 The calculated radial and axial temperature distribution at the steady state.

Fig. 12 Normalized step response temperature, for various positions of thermocouples following the 1% of heat flux increase. The inset gives the temperature increase for 1% power and the step response time.

Fig. 13 Step response time calculated for NILOC I and NILOC II by two-dimensional time-dependent finite difference method. Step response time is shown for the thermocouple TC5 and TC11 positions for different linear power ratings. Power ratings of the NILOC experiments are also shown in the figure. For the two thermocouple positions and for the two experiments the measured step response times with their standard deviation are included.
THERMOHYDRAULICS

Session chairman: P. Liewers (F.R.G.)
SUMMARY OF THE SESSION

In this session four papers are presented devoted to the use of low frequency fluctuations for surveillance and parameter estimation in PWRs. Puyal et al. describe on the one hand some results of monitoring the internal structure of the French 900 MW-and 1300 MW-PWRs by means of ex-core neutron detectors and accelerometers. Comprehensive statistical material is available.

On the other hand investigations of thermohydraulic phenomena are presented, such as:

- temperature fluctuations in the hot legs caused by the use of gadolinium in the core,
- pressure fluctuations due to the pressurizer-loop interaction.

Kostic et al. and Katona and Kozma handle the problems of information in the low frequency fluctuations of incore-neutron and core-exit thermocouple signals. The first paper gives very comprehensive experimental data measured in the NPP Grohnde together with some theoretical interpretation. The second paper concentrates more on the theoretical models and performs comparisons with experimental results from different plants. Thereby it is shown that the usual theoretical approach with the assumption of propagating void contents leads to some inconsistencies. Both papers come to the conclusion that thermohydraulic parameters can be determined by analysing the low frequency fluctuations. In the discussion Wach pointed to the problem that in general mechanical vibrations must be taken into account too, even at low frequencies.

Glockler and Upadhyaya present a multivariate noise analysis method which seems to be very useful for plant diagnosis. It allows the separation of process and sensor anomalies and can be automated. The method was applied to experimental data of the LOFT-reactor and showed convincingly its feasibility of identifying noise sources and signal transmission paths.
USE OF LOW FREQUENCY FLUCTUATIONS FOR THE SURVEILLANCE OF STRUCTURES, SENSORS AND THERMOHYDRAULIC PHENOMENA ON THE 900 MW AND 1300 MW REACTORS

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Electricité de France, Direction des Études et Recherches, 6 Quai Watier, 78400 Chatou, France

ABSTRACT

Low frequency fluctuations observed on the sensors mounted on the PWRs ex core neutron detectors and accelerometers have been used for a long time in order to monitor the internal structures of the reactors. In this paper, results obtained on the new series of 1300 MW reactors will be given : they concern the startup of the reactor Paluel 1 and the monitoring of following units in the last two years. Automatic monitoring of the internals including aid to diagnostic is now under validation on a new series of 12 plants : the system will be briefly described. As far as 900 MW reactors are concerned we will present some results got from our data base.

Low frequency fluctuations also appear on thermodynamic sensors. Three examples are presented :
1 - Gadolinium in the core generates temperature fluctuations in the loops.
2 - Following the oscillations of pressure in the loops may improve the knowledge of fluid state. Experimental results are in a very good agreement with theoretical approach.
3 - Analysis of the fluctuations is an aid to detect impulse line clogging on pressure sensors.

A method based on noise analysis is being experimented on nuclear sites to detect degradation of neutron sensors.

MOTS CLES
Réacteur à eau sous pression, REP 900 MW, REP 1300 MW, PWR, Bruit neutronique, Accéléromètres, Signature spectrale, Densité spectrale de puissance, Détection d'anomalies, Surveillance en service, Vibrations, Fluctuations de température, Fluctuations de pression, Structures internes, Surveillance capteurs, Résonnances acoustiques.

INTRODUCTION

Les fluctuations de basse fréquence et de faible amplitude recueillies sur les capteurs placés autour du cœur des PWR (accéléromètres, chambres neutroniques ex core) sont utilisées depuis plusieurs années pour suivre le comportement mécanique des structures internes 900 MW (réf. 1, 2, 3, 4).

Une nouvelle série de 8 réacteurs 1300 MW venant d'être mise en service, nous présentons les résultats obtenus au démarrage et au cours des deux premiers cycles. La connaissance acquise va être utilisée pour 12 tranches successives, à partir de Cattenom 1, à l'aide d'un nouveau système de surveillance et d'aide au diagnostic.

Pour ce qui concerne le palier 900 MW (34 tranches), quelques éléments tirés de la banque des données du bruit neutronique sont présentés.

D'autres phénomènes de nature thermohydraulique peuvent être appréhendés par une analyse basse fréquence du bruit d'autres capteurs tels que sondes de température et de pression, chambres neutroniques in core. Nous présentons plus particulièrement l'état des travaux effectués sur les fluctuations de pression et sur les fluctuations de température, au travers de quelques exemples visant à montrer l'intérêt de ces techniques.

La surveillance des capteurs eux-mêmes par l'analyse de leur bruit propre est également évoquée en ce qui concerne les capteurs neutroniques.
I - SURVEILLANCE VIBRATOIRE DES STRUCTURES INTERNES

1.1 - Le palier 1300 MW P4 (8 tranches)

Une détection précoce des dégradations pouvant survenir sur les structures internes des réacteurs est essentielle afin d'éviter des indisponibilités prolongées. Le panier de coeur notamment doit faire l'objet d'une surveillance particulière. Il s'agit d'une structure cylindrique suspendue et encastrée en partie haute de cuve à l'aide d'un anneau de calage : cette structure vibre sous l'effet des jets d'eau d'entrée et entraîne le coeur dans son mouvement principal de balancement. Comme pour les réacteurs 900 MW "tête de série" précédents (Fessenheim 1, Tricastin 1), les structures internes de Paluel 1 (premier réacteur 1300 MW) ont été équipées d'accéléromètres lors des essais sans combustible. Les résultats obtenus ont été comparés aux résultats des calculs effectués par le constructeur Framatome : ils servent de point zéro pour la surveillance en exploitation à partir de l'instrumentation externe. Nous allons voir successivement,

- que cette instrumentation est apte à détecter les mouvements principaux des structures internes,
- qu'une surveillance en exploitation des huit premières tranches depuis 1984 permet de mieux connaître les mouvements de structures,
- que d'autres mouvements mécaniques peuvent être détectés à partir de cette instrumentation.

I - Surveillance du panier de coeur à partir des capteurs externes à la cuve

Lors des essais à chaud sans combustible, des accéléromètres ont été montés sur les structures internes inférieures et supérieures de Paluel 1, tête de série 1300 MW (fig. 1). L'analyse croisée entre les accéléromètres permet de définir et d'identifier les principaux modes vibratoires des structures internes.

Sur la figure 2, par exemple, sont reproduits les spectres de cohérence et de phase entre deux accéléromètres A6 et A8 montés de part et d'autre du panier de coeur, et les différents mouvements mis en évidence :

- balancement du panier de coeur entre 8 et 15 Hz,
- mouvement en coque n = 2 du panier de coeur entre 20 et 23 Hz,
- mouvement de balancement du panier de coeur, entraîné par le balancement des structures internes supérieures, entre 30 et 33 Hz.

Cependant, lors de ces essais, nous avons constaté des changements de comportement du panier de coeur. A la figure 3, sont reproduits deux spectres d'un accéléromètre interne monté sur le panier de coeur, à quelques heures d'intervalle : le premier met en évidence un balancement autour de 11 Hz, avec une amplitude de l'ordre de 18 microns efficaces, il s'agit d'un mouvement avec appui sur le bas de la cuve. Le deuxième met en évidence un mouvement autour de 9,1 Hz avec une amplitude de 44 microns efficaces. Il y a cette fois absence de contact avec le bas de la cuve. Un accéléromètre monté sur l'extérieur de la cuve voit bien le mouvement de balancement des structures internes inférieures et supérieures (cohérence entre accéléromètre interne et externe voisine de 0,9 autour de 9 Hz et 31 Hz) : la cuve est mise en mouvement forcé autour de ces fréquences mais l'amplitude est plus faible (2,6 microns efficaces, en absence de contact, à 9 Hz).

Lorsque le réacteur est chargé et en puissance, l'analyse des fluctuations neutroniques issues des chambres ex core montre que ces dernières sont corrélées aux accélérations de la cuve autour des fréquences de balancement du panier de coeur. (voir sur figure 7 le pic autour de 13 Hz sur les DSP et la cohérence entre accéléromètre et bruit neutronique).

Les accéléromètres de cuve et les chambres neutroniques ex core constituent ainsi deux moyens privilégiés et complémentaires pour surveiller les structures internes en exploitation.

2 - Bilan de la surveillance en exploitation des 8 premiers réacteurs 1300 MW

Le système de surveillance des structures internes 1300 MW utilise les sections basse n° 2 et haute n° 5 des quatre chambres ex core multiétagées (à six sections) SB et SH et quatre accéléromètres placés sous la cuve des réacteurs sur les tubes guides de l'instrumentation interne (figure 4). Le système comprend un analyseur de spectre, un multiplexeur de voies et des périphériques pilotés par un calculateur, ainsi que divers logiciels. Ceux-ci sont utilisés,
mensuellement pour détecter des évolutions des signatures de référence effectuées au démarrage,

- trimestriellement pour avoir un état vibratoire complet du circuit primaire.

Tous les relevés effectués sur chaque réacteur sont centralisés à Chatou à la Direction des Etudes et Recherches. A partir des relevés effectués sur les huit tranches Paluel 1 à 4, Flamanville 1,2 et St Alban 1,2, au cours de leurs deux premiers cycles de combustibles (sur douze années - réacteurs de fonctionnement), nous avons pu établir des comportements type, notamment en ce qui concerne le panier de coeur.

Comportement du panier de coeur

A la différence des tranches 900 MW, le panier de coeur est guidé en partie basse par six guides radiaux au lieu de quatre, d'où une plus grande probabilité de contact avec un ou plusieurs d'entre eux, soit lors de l'introduction des internes dans la cuve, soit en fonctionnement. Cela explique les trois types de comportements rencontrés sur les réacteurs 1300 MW (figure 5):

1. Balancement de tendance libre avec appuis intermittents (8 cas/23)
   Cas type Paluel 2 (figure 6)

2. Balancement de tendance appuyée avec libération des contacts intermittents (7 cas/23)
   Cas type Paluel 4 (figure 7 et 8)

3. Balancement très faible car blocage des internes sur plusieurs guides (8 cas/23)
   Cas type Paluel 1 (figure 5c).

Ces trois situations peuvent être caractérisées en examinant les formes des signatures spectrales du bruit neutronique et de l'accélération de la cuve et les fonctions spectrales croisées, entre chambres ex-core d'une part, entre accéléromètres externes et chambres ex-core d'autre part, plus particulièrement entre 7,5 et 15 Hz. Les paramètres de réduction utilisés sont les amplitudes des pics détectés, les écarts types relatifs de la fluctuation neutronique entre 7,5 et 9,5 Hz et entre 9,5 et 15 Hz, la phase entre signaux. Ils sont comparés à des valeurs de référence résultant de l'observation du parc. Le déplacement du panier de coeur est estimé au niveau médian inférieur du coeur par la relation

\[
\frac{d_i \cdot h}{1} = x_{\text{eff}} = \int_{7,5 \text{Hz}}^{15 \text{Hz}} (\text{DSPN}) \cdot \frac{d_i}{i} \cdot df, \text{ où } h \text{ est pris égal à } 0,13 \text{cm}^{-1}
\]

\(\Delta x\) est inférieur à 120 microns efficaces.

D'autre part, au cours d'un même cycle, nous avons pu constater que le panier de coeur pouvait passer d'un type de comportement à un autre.

Ainsi, comme on pouvait s'y attendre lors des essais à chaud de Paluel 1, le comportement vibratoire des structures internes est complexe et sa connaissance nécessite le recours à un personnel spécialisé. Un système de traitement spécifique peut cependant aider à l'interprétation, comme c'est le cas sur le palier 1300 MW (P4).

3 - Autres mouvements accessibles à partir de l'instrumentation de surveillance externe à la cuve

En dehors de la zone 7,5 à 15 Hz correspondant au balancement du panier de coeur, le bruit neutronique perçoi,

- les mouvements poutre n = 1 et n = 2 des assemblages combustibles autour de 2,5 Hz et 5,8 Hz.
- les mouvements en coque n = 2 de l'enveloppe de coeur autour de 20 à 22 Hz.

Les accéléromètres voient nettement le balancement de la cuve autour de 17 à 18 Hz ainsi que le balancement des structures internes supérieures autour de 32 Hz.

4 - Conclusions

La figure 9 rassemble les résultats des essais de démarrage et de la surveillance du parc 1300 MW. Elle indique les fréquences observées des modes propres pour les fréquences surveillées ainsi que les capteurs utiles à la détection : ces fréquences sont en excellent accord avec celles calculées par le constructeur.
I.2 - Le palier 1300 MW P4 (12 tranches)

La connaissance acquise sur les structures internes 1300 MW du palier P4, identiques à celles du palier P4 d'une part, les évolutions technologiques, d'autre part, ont permis d'ajouter une fonction d'aide au diagnostic au nouveau système de surveillance vibratoire des douze prochaines tranches 1300 MW (palier P4).

Ce système est en place à Cattenom 1 et 2 depuis septembre 86. Il comporte un multiplexeur, un analyseur de spectre à deux voies, un calculateur, un disque dur, et des périphériques (figure 10). Plusieurs logiciels permettent :

- la détection d'anomalies sur les spectres de bruit neutronique, par comparaison aux spectres de référence établis lors du démarrage - le contrôle est mensuel et activé à la demande.

- l'analyse vibratoire de chaque voie séparément - le contrôle est trimestriel et activé à la demande.

- l'analyse vibratoire croisée entre voies et le prédagnostic - le contrôle est biannuel avant et après redémarrage et il est activé à la demande.

Dans cette configuration, le système utilise l'accéléromètre de cuve et huit sections basse et haute des chambres neutroniques. Il identifie les mouvements de structures internes, le comportement du combustible et détecte des anomalies vibratoires et en réactivité. Il quantifie les signaux observés et stocke les états de référence, permet une comparaison des situations dans le temps (figure 11).

- la surveillance en continu des signatures spectrales, globalement et par bandes de fréquence.

Ainsi il sera possible d'identifier directement sur site, le type de comportement des structures et d'interpréter si l'évolution constatée est liée à une anomalie d'origine vibratoire ou à une modification normale. La tâche des experts devrait en être facilitée.

I.3 - L'état de la surveillance du parc actuel 900 MW

La surveillance des 45 tranches actuellement en service a commencé dès leur démarrage en 1976. Elle a permis de constituer une banque de données de bruit neutronique qui couvre actuellement plus de 100 années réacteurs. Cette banque comprend (Réf. 4),

- les écarts types du signal fluctuant par bande de fréquence, chaque bande correspondant à un phénomène ou un mouvement de structure différent.

- les pics principaux du spectre caractérisés chacun par sa DSPmax, sa fréquence centrale et son amortissement.

Nous avons pu évaluer ainsi l'effet sur le spectre de l'usure du combustible : l'étude a porté sur 24 réacteurs du type écran thermique circulaire ou en secteur. Le tableau ci-dessous donne pour chaque type de réacteur les zones de fréquence étudiées, leur identification et l'accroissement de l'écart type du bruit neutronique sur les sections basses des chambres ex-core, au cours d'un même cycle.

<table>
<thead>
<tr>
<th>Phénomène surveillé</th>
<th>Gamme de fréquences Hz</th>
<th>Évolution de l'écart type du bruit en cours de cycle</th>
</tr>
</thead>
<tbody>
<tr>
<td></td>
<td>Écran circulaire(6)</td>
<td>Écran en secteurs(28)</td>
</tr>
<tr>
<td>Tous phénomènes</td>
<td>0 à 24</td>
<td>+ 0,35</td>
</tr>
<tr>
<td></td>
<td></td>
<td>+ 0,6</td>
</tr>
<tr>
<td>Thermohydraulique</td>
<td>0 à 2</td>
<td>+ 0,38 à + 0,4</td>
</tr>
<tr>
<td></td>
<td></td>
<td>+ 0,47 à + 0,52</td>
</tr>
<tr>
<td>Combustible</td>
<td>Mode 1</td>
<td>+ 0,5</td>
</tr>
<tr>
<td></td>
<td>0 à 5</td>
<td>+ 0,25</td>
</tr>
<tr>
<td></td>
<td>5 à 6,8</td>
<td></td>
</tr>
<tr>
<td></td>
<td>6,8 à 10</td>
<td>+ 0,91</td>
</tr>
<tr>
<td></td>
<td>7,5 à 10</td>
<td></td>
</tr>
<tr>
<td>Panier de coeur</td>
<td>Mode coque n°2</td>
<td>+ 0,55</td>
</tr>
<tr>
<td></td>
<td>10 à 12</td>
<td></td>
</tr>
<tr>
<td></td>
<td>12 à 16</td>
<td>+ 0,41</td>
</tr>
<tr>
<td></td>
<td>Mode coque n°3</td>
<td></td>
</tr>
<tr>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td></td>
<td>Panier de coeur</td>
<td>+ 0,26</td>
</tr>
<tr>
<td></td>
<td>Mode coque n°2</td>
<td></td>
</tr>
<tr>
<td></td>
<td>18 à 24</td>
<td>+ 0,20</td>
</tr>
</tbody>
</table>

() nombre de tranches

Tableau n° 1 : Identification des zones de fréquences entre 0 et 24 Hz
Cette augmentation varie selon les gammes entre 20 % et 60 %. Elle est liée au durcissement du spectre des neutrons de fission par production de Pu 239, ce qui accroît le coefficient $h$ précédemment défini. Un calcul reste à faire sur l'évolution de ce coefficient $h$.

A partir de cette étude nous préparons la mise en place d'un système de surveillance complètement automatisé sur les tranches 900 MW. Cependant, le besoin n'est ressenti que sur les tranches les plus anciennes, pour lesquelles un vieillissement des structures pourrait intervenir et des défauts apparaître ; il faut noter en effet que le comportement des structures internes s'est montré excellent au cours des dix dernières années.

II-ANALYSE DE PHÉNOMÈNES THERMOHYDRAULIQUES

L'analyse des fluctuations transmises par les capteurs de pression et température permet la détection et la caractérisation de certaines anomalies, imputables soit à la grandeur physique, soit au capteur lui-même. Les quelques cas présentés montrent les possibilités d'application de la méthode à la surveillance.

II.1 - Fluctuations de température des branches chaudes (Gravelines 5)

Au début de 1985, des fluctuations de température anormalement élevées sont observées sur les branches chaudes de Gravelines 5, de l'ordre de 0,6°C crête à crête au lieu de 0,3 à 0,4°C généralement constaté sur des réacteurs de ce type. Cela peut entraîner un arrêt d'urgence du fait d'un écart température moyen/consigne trop important. A la demande de la centrale, une investigation a donc été effectuée sur ce réacteur. Une autre investigation a eu lieu sur un deuxième réacteur, Cruas 2, à partir de mêmes capteurs.
Le bruit des capteurs est obtenu après rejet de la composante continue et amplification. Il est caractérisé par sa valeur RMS et par son spectre de fréquence.

On constate (tableau 2) que la fluctuation est de l'ordre de 25 % plus élevée sur Gravelines que sur Cruas (voir figure 12). Ces fluctuations ont été corrélatées par ailleurs avec celles observées sur d'autres capteurs de boucle, débit, pression, flux neutronique. Les cohérences étant très faibles, il ne pouvait s'agir de phénomènes intéressant l'ensemble du réacteur (figure 13).
La raison de ces fluctuations a pu être expliquée par la présence de gadolinium dans le coeur. En effet, les discontinuités de chargement entraînent, à la sortie des éléments combustibles, des écarts de température entre filets d'eau plus importants que sur un autre réacteur.
Ces écarts se retrouvent au niveau des températures en branche chaude.

La solution a alors consisté à intervenir sur l'électronique de mesure pour éviter un éventuel déclenchement de la tranche.

![Tableau 2](image)

**Tableau 2 : Fluctuations de températures efficaces observées sur les branches froides et chaudes des réacteurs.**

II.2 - Les oscillations de pression des boucles primaires

Différents travaux ont montré la possibilité de relier les oscillations de pression basse fréquence des boucles primaires à des états thermohydrauliques différents du fluide (Références 5, 6, 7). L'étude réalisée à Electricité de France a eu pour but de mettre en évidence ce phénomène sur réacteur et à l'évaluer en vue d'une application éventuelle (référence 8).

Nous sommes partis d'une modélisation du circuit primaire par un circuit acoustique simple, dans lequel interviennent les volumes d'eau et de vapeur dans la cuve et les boucles primaires (V4 et V3 respectivement), dans le pressuriseur (V2 et V1 respectivement) et dans la ligne d'expansion ($Q_S = longueur \times section$).
On peut alors, par analogie électrique du modèle acoustique simple, définir une capacitance $C_i = \frac{V_i}{\rho_i a_i^2}$ et une inerance $M = \frac{\rho_i}{S}$.

Expressions dans lesquelles

$V_i =$ volume du fluide $i$

$a_i =$ vitesse du son dans le milieu $i$

$\rho_i =$ densité du fluide $i$

La fréquence de résonance de ce circuit est $F_{res} = \frac{1}{2\pi} \sqrt{\frac{1}{M \cdot C_{eq}}}$

avec $C_{eq} = \frac{(C_1 + C_2)(C_3 + C_4)}{C_1 + C_2 + C_3 + C_4}$

On peut alors utiliser ce modèle pour étudier trois situations.

1. Le pressuriseur est solide (pression 30 bars, température inférieure à 220° C) il n'y a pas de vapeur dans la cuve $C_3 = 0$, et $C_1 = 0$

   $F_{res} = \frac{1}{2\pi} \sqrt{\frac{C_2 + C_4}{M \cdot C_2 \cdot C_4}}$

2. Le fonctionnement est normal (pression 155 bars, température moyenne coeur entre 280° et 300° C) il n'y a pas de vapeur dans la cuve $C_3 = 0$ $F_{res} = \frac{1}{2\pi} \sqrt{\frac{C_1 + C_2 + C_3}{M(C_1 + C_2) \cdot C_4}}$

3. Il y a ébullition dans le coeur (pression 155 bars, température d'eau primaire 345° C

   $F_{res} = \frac{1}{2\pi} \sqrt{\frac{C_1 + C_2 + C_3 + C_4}{M(C_1 + C_2)(C_3 + C_4)}}$

Cette modélisation a été appliquée à trois réacteurs, Tihange 1, Triscatin 1, ainsi que sur Paluel 1 et elle a été vérifiée expérimentalement (figures n° 14 et n° 15). Les résultats sont reportés dans le tableau n° 3 ci-après.

L'accord théorie-experience est très satisfaisant.

Ainsi, on voit que la première fréquence propre acoustique observée sur le bruit de pression est imputable à l'état du fluide primaire et à la présence ou non de vapeur au pressuriseur ou dans les boucles.

Le suivi de cette fréquence permet donc d'envisager une utilisation pour contrôler certains paramètres de fonctionnement du réacteur.
a) Mesure du taux de vide au pressuriseur (figure 15,a)  
Cette figure donne aux conditions nominales la variation de la fréquence de résonance avec le taux de vide au pressuriseur. On voit que celle-ci passe de 0,68 Hz pour un taux de vide de 40 % à 1,02 Hz lorsque le pressuriseur est solide.

<table>
<thead>
<tr>
<th>Pressuriseur solide</th>
<th>Fréquence théorique (Hz)</th>
<th>Fréquence expérimentale (Hz)</th>
</tr>
</thead>
<tbody>
<tr>
<td>(30 bars, 70°C)</td>
<td></td>
<td></td>
</tr>
<tr>
<td>Tricastin</td>
<td>2,38</td>
<td>Non vérifié</td>
</tr>
<tr>
<td>Tihange (fig.14)</td>
<td>1,92</td>
<td>1,87</td>
</tr>
<tr>
<td>Paluel</td>
<td>1,57</td>
<td>Non vérifié</td>
</tr>
<tr>
<td>Conditions nominales</td>
<td></td>
<td></td>
</tr>
<tr>
<td>(155 bars, 345°C)</td>
<td></td>
<td></td>
</tr>
<tr>
<td>Tricastin (fig.14)</td>
<td>0,677</td>
<td>0,675</td>
</tr>
<tr>
<td>Tihange (fig.14)</td>
<td>0,547</td>
<td>0,575</td>
</tr>
<tr>
<td>Paluel</td>
<td>0,443</td>
<td>Non vérifié</td>
</tr>
</tbody>
</table>

Tableau 3 : Modélisation du circuit primaire  
Résultats théoriques et expérimentaux

II.3 - Dégradation de la réponse dynamique d'un capteur de pression

En février 1987, la centrale de Gravelines 5 demande une investigation sur un capteur de pression placé sur la turbine. Ce capteur a pour fonction de déclencher le circuit de contournement turbine, lors de transitoires de pression de grande amplitude. Tout retard de réponse du capteur supérieur à 0,9 seconde peut, en effet, entraîner un arrêt d'urgence. Or le capteur ne semble pas répondre rapidement et un contrôle en statique n'a rien donné.

L'analyse de bruit est effectuée avant arrêt pour rechargement. Elle peut fort heureusement être comparée à une analyse similaire faite auparavant, en 1986. La superposition des spectres montre en février 1987 (figure 16) une absence totale de fluctuation dans la bande de fréquence utile (10 Hz). Ce qui permet de conclure à une obturation de la ligne d'impulsion. Le bouchage n'étant pas étanche, la pression se transmet lentement tandis que les transitoires sont filtrés, ce qui augmente le temps de réponse du système. Ces résultats ont été confirmés lors de l'arrêt suivant par un examen du tube de transmission. Lequel a été trouvé obturé.

Cet exemple montre l'intérêt de la méthode. Cependant elle nécessite de disposer d'un catalogue de signatures de référence et elle est jugée d'application délicate par des non spécialistes.

III - LA SURVEILLANCE DES CAPTEURS NEUTRONIQUES

Une politique de remplacement des capteurs neutroniques est en train de se définir sur les tranches nucléaires d'EDF ; deux méthodes sont utilisées :
- la première, est basée sur l'étude de la déformation des courbes de saturation courant - haute tension,
- la deuxième est basée sur l'analyse du bruit de détection de la chambre (figure 17).

Cette deuxième méthode déjà présentée à SMORN IV (référence 9) consiste à relier un descripteur de ce bruit de détection à la dégradation de la chambre. La méthode est en application sur plusieurs reacteurs (Fessenheim, Cruas,...) avec des chambres jugées douteuses. Elle ne sera jugée intéressante que si elle s'avère plus sensible que la précédente. Pour l'instant, nous ne disposons pas encore de résultats significatifs.
IV - CONCLUSIONS

Les fluctuations de basse fréquence observées sur les capteurs d'exploitation montés sur les circuits primaire ou secondaire des réacteurs nucléaires, peuvent dans certains cas constituer une gêne pour la mesure de la grandeur moyenne, mais elles peuvent apporter beaucoup à la connaissance du fonctionnement des réacteurs.

La surveillance des structures internes est ainsi assurée systématiquement à partir des fluctuations neutroniques des chambres ex core, et des accélérations de la cuve. Le retour de l'expérience acquise sur le parc 900 MW et 1300MW permet à présent d'envisager une plus grande automatisation et une décentralisation du diagnostic sur les sites.

Les fluctuations de pression et de température pourraient cependant faire l'objet d'une surveillance accrue, compte tenu de l'intérêt qu'elles présentent soit pour améliorer la maintenance de l'instrumentation, soit pour mieux connaître l'état thermodynamique du fluide. Des efforts devraient être faits dans ce sens au cours des prochaines années.

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**Figure 1 :** Accéléromètres montés sur les structures internes et sur la cuve lors des essais à chaud de Paluel 1 (1983)

**Figure 2 :** Cohérence et phase entre A6 et A8 lors des essais à chaud

**Figure 3 :** Évolution de signature au cours d'essai à chaud et corrélation entre capteur interne et externe à la cuve.

**Figure 4 :** Instrumentation de surveillance en exploitation des structures internes 1300 MW.
Figure 5 : signatures types observées sur les réacteurs 1300MW
a) mode libre (appui intermittent)
b) mode avec appui prépondérant
c) mode avec appuis multiples

Figure 6 : a) balancement en mode libre prépondérant (PA2)

Figure 7 : b) balancement avec appui prépondérant (PA1)

Figure 8 : b) balancement avec appuis multiples (PA4)
Figure 9 : caractérisation des modes vibratoires des structures internes 1300 MW par bruit neutronique et accéléromètres

Figure 10 : réacteurs 1300 MW P'a
Système de surveillance vibratoire et d'aide au diagnostic

Figure 11 : exemple de relevé effectué en automatique sur Cattenom 1.
Figure 12 : spectre et écart type des fluctuations de température en branche chaude de Gravelines 5 et Cruas 2

Figure 13 : cohérence entre le signal de température et de débit d'une boucle chaude de Gravelines 5

Figure 14 : spectres de pression dans les boucles de circuit primaire

Figure 15 : a) variation de la fréquence de résonance avec le taux de vide au pressuriseur

b) variation de la fréquence de résonance avec le taux de vide dans la cuve
Figure 16 : Gravelines 5
Dégradation de la réponse dynamique d'un capteur
par obturation de la ligne d'impulsion

Figure 17 : Descripteur du bruit de détection
THERMOHYDRAULICS SURVEILLANCE OF PRESSURIZED WATER REACTORS BY EXPERIMENTAL AND THEORETICAL INVESTIGATIONS OF THE LOW FREQUENCY NOISE FIELD

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ABSTRACT

Experimental and theoretical investigations of the neutron and temperature noise field have been performed in the nuclear power plant Grohnde (PWR, 1350 MW). Noise measurements under different operating conditions during several fuel cycles were made with long recording times. Evaluations in the low frequency range succeeded in measuring the coolant velocity and in detecting subcooled boiling. This is important for determination and localization of essential deviations and possible anomalies like in-core flow blockages, localized power skews, hot spots etc.

KEYWORDS

PWR noise, low frequencies, thermohydraulics, coolant velocity, subcooled boiling

1. INTRODUCTION

Theoretical investigations of the noise in the last few years show that the power noise can be described not only by the point reactor model. In the point kinetic approximation of the neutron noise theory, the fluctuation group parameters do not drive the noise directly but rather via fluctuations of the reactivity of the core.

According to the space dependent theory, however, the reactivity induced noise is only one component of the noise field. There also exists another component that represents the direct local influence of the fluctuating group constants. It is resonable to call the reactivity induced fluctuations the global component of the noise, whereas the rest of the noise may be called the local component.

Many experimental works published in the last few years show the importance of the space dependent effects, too. The main effort of these investigations concentrated on the frequency range above 1 Hz. Low frequencies coupled with thermal hydraulic neutron kinetic phenomena were investigated in a few papers. In the low frequency range, both feedback effects and different noise sources become active.

PNE-O
In a PWR, the noise analysis must be carried out by investigating the effects of four kinds of perturbance sources on the neutron flux and core-exit temperature dynamics: coolant flow rate fluctuations, random heat transfer fluctuations due to turbulence, coolant core inlet temperature fluctuations and heat source fluctuations. The heat source fluctuations include all the disturbance sources that affect the neutron flux first and then through heat transfer processes cause the fluctuation of the core-exit temperature.

A local/global model was published some years ago (Wach and Kosaly, 1974) for the interpretation of the noise field in a BWR. According to this model, the local component of the neutron noise in a PWR at a given space point is directly proportional to the fluctuation of the coolant temperature (Katona et al, 1982), while the global component can be described by the point reactor model, that is, it can be considered to be driven by reactivity fluctuations.

The space dependent model (Kozma and Mesko, 1985) has a somewhat intermediate character between the point model and the methods of spatial higher harmonics. In this paper, authors show how the thermal hydraulic feedback affects the spectral characteristics of a power reactor.

In the paper (Kozma, 1985) the effect of the temperature feedback on the eigenvalues of the neutron noise components is examined using a one dimensional homogeneous reactor noise model. It is shown, that at low frequencies (below 1-2 Hz) strong feedback effects influence the eigenvalue associated with the global component of the neutron noise. Namely, this global eigenvalue splits into three parts. At the same time the eigenvalue associated with the local component remains unaffected.

This actual paper presents noise measurements at the 1350 MW PWR Grohnde (KWG) in the low frequency range. The aim of measurements was to determine the noise patterns of the reactor as completely as possible, to monitor these patterns and to interpret them physically as far as possible.

2. THEORETICAL MODEL

The axial dependent model starts with the physical assumption that the fluctuations of the neutron density at a given space point inside the core is composed of two parts: a local part driven by local disturbances and a global one representing a more or less coupled behaviour of the neutron field.

For the cross spectral calculations, we assume that the thermocouple signal fluctuation at the position \( z \) is proportional to the local coolant temperature fluctuations.

\[
S_t(z, \omega) = \varepsilon \ast S_C(z, \omega)
\]  

(1)
In-core neutron signal is a sum of two terms, one is proportional to the reactor power (global component), and the other (local component) to local sources as temperature fluctuations.

\[ S_1(z, \omega) = g \delta P(\omega) + \delta T_C(z, \omega) \tag{2} \]

In the above relation, the neutron noise field is coupled to the temperature through the local and global effects.

The fluctuations of the ex-core ionization chamber is proportional to the reactor power,

\[ S_{ex}(\omega) = \delta P(\omega) \tag{3} \]

what means that the ex-core detector "sees" the global noise only.

Supposed (Runkel, 1987) that the independent noise is the dominant noise source in the core and that the inlet fluctuations can be neglected, then the coolant temperature fluctuation at the position \( z \) can be given by

\[ T_C(z, \omega) = |\delta T_C(z, \omega)| \cdot \delta P(\omega) \cdot \exp(-i\theta_C(z, \omega)) \tag{4} \]

with

\[ |\delta T_C(z, \omega)| = \frac{q}{v\tau_c} \cdot \frac{z}{(1 + (2w^2) + \frac{2}{\alpha})} \cdot \sin(\omega z/2v) \tag{4a} \]

\[ \delta P(\omega) = P_{in} \cdot G(\omega) \tag{4b} \]

\[ \theta_C(z, \omega) = \frac{z\omega}{2v} + \arctg(\omega\tau_f) \tag{4c} \]

The cross power spectral density function between two signals is:

\[ \text{CPSD}_{ij}(z_i, z_j, \omega) = S_i(z_i, \omega) * S_j^*(z_j, \omega) \tag{5} \]

where \( S_j(z_j, \omega) \) is the complex conjugate value of the signal \( S_j(z_j, \omega) \).

Introducing signals from Eqs. 1, 2, 3 into Eq. 5, different cross power spectral density functions can be obtained.

Accordingly, the cross power spectrum between in-core and thermocouple signals is:

\[ \text{CPSD}_{i-t}(z_i, z_t, \omega) = e g \delta P(\omega) \cdot |\delta T_C(z_t, \omega)| \cdot \exp(-i\theta_{i-t}(z_i, z_t, \omega)) \tag{6} \]

where \( Y_i \), a quantity characterizing the relative importance of local to global components, is the so-called peaking factor in the core (Behringer et al., 1977)

\[ Y_i = \frac{1}{g} \cdot |\delta T_C(z, \omega)| = \frac{\delta L(z, \omega)}{S G(z, \omega)} \tag{6a} \]

and

\[ \theta_{i-t}(z_i, z_t, \omega) = \frac{\omega}{2v} (z_t - z_i) \cdot \arctg \frac{\sin(\omega z_i/2v - \arctg(\omega\tau_f))}{Y_i \cdot \cos(\omega z_i/2v - \arctg(\omega\tau_f))} \tag{6b} \]
The cross power spectral density function between two in-core neutron detector-signal at different axial positions is:

\[
\text{CPSD}_{i-1}(z_1, z_2, \omega) = g^2 |S P(\omega)|^2 |\mathcal{Y}| \exp[-i \theta_{i-1}(z_1, z_2, \omega)]
\]  \hspace{1cm} (7)

Here

\[
|\mathcal{Y}| = \{|(\cos(\theta_1 - \theta_2) - Y_1 \cos \theta_2 Y_2 \cos \theta_1) + Y_1 Y_2|^2 + (\sin(\theta_1 - \theta_2) - Y_1 \sin \theta_2 Y_2 \sin \theta_1|^2)\}^{1/2}
\]

In the case that the local effect is dominant, \(Y > 1\), (i.e., hot channel), the phase between neutron detectors and thermocouple (Eq. 6) is linear and equal to

\[
\theta_{i-1}(z_1, z_2, \omega) = \theta_{i-1}(z_2 - z_1, \omega) = \frac{\omega}{2v}(z_2 - z_1) = \theta_{i-1}(z_1, z_2, \omega)
\]  \hspace{1cm} (8)

When the global effect is dominant

\[
\theta_{i-1}(z_1, z_2, \omega) = \arctg\left[\frac{\omega z_1}{2v} \cdot \arctg(\omega \tau_f)\right] = \frac{\omega z_1}{2v} \cdot \arctg(\omega \tau_f)
\]

Therefore, the phase \(\theta_{i-1}(z_1, z_2, \omega)\) is equal to

\[
\theta_{i-1}(z_1, z_2, \omega) = \frac{\omega z_1}{2v} \cdot \arctg(\omega \tau_f)
\]

and does not depend on the neutron detector position in the core.

When \(L \approx G, Y \approx 1\), the linear phase is modulated with the second term.

Supposed that the local to global ratio is space independent (\(Y = Y_2^2 Y\)). Then the phase between two in-core neutron signals (Eq. 7) is

\[
\theta_{i-1}(z_1, z_2, \omega) = \frac{\omega}{2v} (z_2 - z_1) \cdot \arctg \left( \frac{\sin(\theta_1 - \theta_2) + (\sin \theta_1 - \sin \theta_2) Y}{Y^2 \cos(\theta_1 - \theta_2) + (\cos \theta_1 - \cos \theta_2) Y} \right)
\]  \hspace{1cm} (11)

If the global effect is dominant, \(Y = 0\), eq. 12 is valid.

\[
\theta_{i-1}(z_1, z_2, \omega) = \frac{\omega}{2v} (z_2 - z_1) + \theta_1 - \theta_2 = 0
\]  \hspace{1cm} (12)

If the local effect is dominant, \(Y = 1\), the phase is described by Eq. 13.

\[
\theta_{i-1}(z_1, z_2, \omega) = \frac{\omega}{2v} (z_2 - z_1)
\]  \hspace{1cm} (13)

If the local to global ratio is near one,

\[
\theta_{i-1}(z_1, z_2, \omega) = \frac{\omega}{2v} (z_2 - z_1) \cdot \arctg \left( \frac{\sin(\theta_1 - \theta_2) + \sin \theta_1 - \sin \theta_2}{1 + \cos(\theta_1 - \theta_2) + \cos \theta_1 - \cos \theta_2} \right)
\]  \hspace{1cm} (14)

the linear phase is modulated with the second term of Eq. 14. Its value is greater than \(-\pi/2\) and less than \(\pi/2\).

3. CORE AND EXPERIMENTAL SET-UP

In-core self powered neutron detector (SPND), ex-core ionisation chamber and core exit thermocouple noise signals at the 1350 MW PWR Grohnde (KWG) were recorded and analyzed at different operating conditions.
In-core neutron detectors were located at different axial and radial positions. The core-exit thermocouples were located at the top of the core.

The schematic lay-out of the core and the locations of the measuring positions are indicated in Fig. 1.

![Fig. 1: Core lay-out and measuring positions](image)

The measurements were performed at full power and at stationary thermohydraulic conditions.

At the end of the fuel cycles core life was extended by the so-called "stretch-out operation". In this mode, reactor boron concentration is zero. At nearly constant primary pressure reactor power is reduced from 100 % nominal power down to 60 % (at a rate of about 0.5 % per day) with temperature falling gradually.

The noise data coming from neutron detectors and thermocouples were recorded for about 120 minutes.

4. EXPERIMENTAL RESULTS

4.1 Noise behaviour in the region 0.1 - 0.8 Hz (full power)

In-core neutron detector/thermocouple

- The maximum coherence is about 0.6.
- The phase is linear for all thermocouples and different in-core neutron detector locations (Fig. 2).
- The coherence does not depend on axial location of neutron detectors (Fig. 2).
- The coherence depends on the radial channel position in the core (Fig. 3).
- The slope of the phase depends on axial but not on radial position of the neutron detectors (Fig. 3).
In-core/in-core detectors (0.1 - 0.8 Hz)

- The NAPSDs of axially located SPND signals do not show significant increase with increasing detector position (Fig. 4).
- NAPSD increases during the fuel cycle (Fig. 5).
- The coherence function between two axially neighboured SPNDs is very high (near one) for different detector locations (Fig. 6).
- The phase between two axially separated detectors shows linear behaviour.
- The slope of the phase increases with increasing distance between detectors.
- The coherence function between in-core detectors in the same horizontal plane decreases with increasing axial core position (Fig. 7).
- The phase between two horizontally separated detectors is equal \( \pm 180^\circ \).

Fig. 2: Correlation between SPNDs and core-exit thermocouple at core position 0 05

Fig. 3: Correlation between SPND 5 and core-exit thermocouple at core positions C 04 and J 02

Fig. 4: NAPSD dependence on radial and axial positions
Fig. 5: Change of the NAPSD (det. 1) during the fuel cycle

Fig. 6: Change of the coherence and phase between SPNDs in the position 0 05 during the fuel cycle

Fig. 7: Radial correlation between SPNDs
4.2 Noise behaviour in the region 0.8 - 1.5 Hz

- The NAPSDs decrease with increasing axial detector locations (Fig. 4).
- The phase does not show linear behaviour in this region (Fig. 6).
- The coherence between axially separated in-core detectors shows sink behaviour. It decreases with the distance between detectors.
- In the lower part of the core there is no sink behaviour.
- The sink behaviour is present in the ex-core neutron signals, too. It does not depend on the ex-core detector position.
- The sink frequency value from the ex-core measurement is in good agreement with the in-core value.
- There is no coherence between in-core detectors and thermocouples.

4.3 Noise behaviour during stretch-out operation

- The phase between in-core detectors is not exactly linear (Fig. 8).
- The coherence between SPNDs is very high and does not show sinks.
- The coherence between in-core detectors and ex-core thermocouples shows linear behaviour and does not depend on the operating conditions (Fig. 9).
- The NAPSD in the upper part of the core depends strongly on the operating condition (Fig. 10).

![Graphs showing coherence and phase](image)

**Fig. 8:** SPNDs phase for different operation conditions
Fig. 9: Change of the phase SPND 5/thermocouple (J 06)

Fig. 10: NAPSD (det. 1) for different operation conditions

5. DISCUSSION AND CONCLUSIONS

The low frequency noise is caused by coolant temperature fluctuations. The change of the noise level with boron concentration is due to the change in temperature coefficient of the core. The linear phase between neutron/neutron detectors observed at full power depends on reactor and fuel assembly power. The linear neutron/temperature phase does not depend on the operating conditions.

From the linear slope of the neutron/neutron and neutron/temperature phase and known distances between neutron detectors and thermocouples, the velocity of the propagating disturbances could be determined (Fig. 11). This velocity is in good agreement with the hydraulic core design value for the coolant velocity.

In-core and ex-core auto and cross correlation noise measurements show sink behaviour at full power at a frequency proportional to the transport time of the coolant through the core.

Axially decreasing NAPSDs of noise signals of SPNDs belonging to the same string can be explained by the axially decreasing coolant density fluctuation within the given fuel assembly.

The axial space dependence of the horizontal coherence means that the coolant density fluctuation (relatively weak and uncorrelated at the top) will become stronger and more correlated between the different assemblies at lower levels. The effects of subcooled boiling to the neutron noise appear to have been observed in the upper part of the core.
Fig. 11: Coolant velocity from the SPND/SPND correlation

In modern PWRs subcooled boiling is allowed at normal operating conditions and thermohydraulic surveillance of the primary system is of growing interest.

Measurement of the coolant velocity and detection of the local subcooled boiling in the reactor core are very important from the point of view of localizing essential deviations and possible anomalies as in-core flow blockages, localized power skews, hot spots etc.

Experimental and theoretical results show that neutron/neutron and neutron/thermocouple phases as well as auto power and cross power neutron spectral density sink frequencies are useful for monitoring core flow in PWRs. The interpretation of the coolant velocities from these signals depends on the dominant noise sources of temperature noise, the response times of thermocouples, locations of the detectors and reactor dynamics.

The space dependent neutron noise effects caused by the coolant boiling were observed in experiments.

It is possible to detect subcooled boiling in a PWR via neutron noise signals.

**ACKNOWLEDGEMENTS**

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PROBLEMS OF ESTIMATION OF THE THERMOHYDRAULIC PARAMETERS USING NEUTRON AND TEMPERATURE NOISE SIGNALS IN PWRs

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ABSTRACT- In the paper the characteristic sinks in the Cross Power Spectral Density (CPSD) and coherency functions of neutron and temperature noise in PWRs are studied. The experimental results and theoretical models for the interpretation of the sink structure are discussed from the point of view of coolant velocity and subcooled boiling determination. A one-dimensional coupled thermohydraulic neutronic reactor noise model is applied for the better understanding of the sink - structure of the characteristic functions of the neutron and temperature noise in PWRs.

1. INTRODUCTION

During the last few years several attempts were made to evaluate the thermohydraulic properties of the PWR core using neutron and temperature noise analysis. This subject is related on one hand to the measurement of the flow velocity in the core and on the other hand to the detection of subcooled boiling. These tasks of the PWR diagnostics are of interest because of possible anomalies such as flow channel blockages, localized power skews, hot spots and departure from nucleate boiling (see Bernard et al., 1986).

In several measurements the Cross Power Spectral Density (CPSD) and the coherency functions (COH) between the core-exit temperature and in-core neutron noise signals and also CPSD between in-core neutron noise signal pairs in PWRs show periodically appearing minima, so called, sinks (see e.g. Bastl et al, 1985, Grondey et al., 1985, Upahyaya and Türkcan, 1984, Lehman et al., 1985). Analysing these experimental results it seems to be possible to calculate the flow velocity in the core from the sink frequencies of CPSD or COH using simple algebraic formulas.

Some of authors identify the propagating perturbances with the void fluctuations in the core and connect the presence of the sinks in CPSD or COH with existence of subcooled boiling in the PWR core (see e.g. Grondey et al., 1985). Usually simple theoretical models are applied for the interpretation of the sink structure of CPSD and COH and the relationship between coolant velocity and sink frequency. These theoretical models developed first for BWR noise analysis are more or less simplified descriptions of the neutronic responses of the core to the propagating perturbances.

In this paper a comparison of measured and theoretically obtained noise characteristics is given. As it will be shown, the usual theoretical approach leads to inconsequences between the theory and experimental results.
Appreciating the obvious simplicity of the abovementioned theoretical treatment we propose a more sophisticated model for the better understanding of the sink-structure of CPSD and COH of neutron and temperature noise in PWRs. The one-dimensional coupled thermo-hydraulic-neutronic model is based on the two-group diffusion equations and complete set of conservation equations for both single-phase and two-phase flow. Using this model the character of the measured Auto- and Cross Power Spectral Density Functions, coherency functions and phases can be reproduced and clarified in the calculations as well. The behaviour of the phase of CPSD between neutron and temperature noise signals was studied earlier by the authors, Kozma, Katona, 1986.

2. THEORETICAL BACKGROUND

In the following chapter we will give a review of the most important theoretical models for the interpretation of the sink structure of reactor noise characteristics functions (APSD, CPSD, COH). We will consider the development of theoretical findings not in a chronological but in a logical way. In the late sixties many authors tried to describe the neutronic response of the core to the propagating perturbances in point kinetic approach. In several calculations the characteristic functions of the noise or the transfer function itselfs has shown dips which appear periodically (see e.g. Robinson, 1967). The first work consciously pointing out the existence of periodically appearing dips in the characteristic functions of the neutron noise caused by propagating perturbances was published by Mogilner (1971). From our point of view an important step was made by Kosály and Meskó (1972). They described the core transfer function relating inlet temperature fluctuation to neutron noise. In this paper the possibility of the determination of coolant velocity based on the sink frequency of neutron noise spectra is shown. The neutronic transfer function is obtained in that work using the point kinetic model, too, but the driving source - i.e. temperature noise in the core due to inlet temperature fluctuation - is calculated using a more sophisticated two-temperature model.

The later development in this field we will consider in two different aspects:
- description of the thermo-hydraulic noise source,
- development of the neutronic models.

2.1. Noise source in the reactor channel in PWRs

Under typical PWR operational conditions one phase flow heat transfer exists in the reactor channel. However, in some advanced PWRs there are hot channels in which subcooled boiling can appear during normal operation too. We consider the thermo-hydraulic noise sources in the PWR core both in one-phase flow and subcooled boiling two-phase flow.

2.1.1. One-phase flow in the reactor

In the case of one-phase flow the coolant is to be characterized by the temperature (or enthalpy), pressure and velocity. The equations of small fluctuations of these state variables can be derived from the conservation equation of mass, momentum and energy. If the reactor channel considered to be one-dimensional flow-channel, these equations are first order partial differential equations with respect to the time and axial coordinate. The equation describing the temperature fluctuations of the fuel can be derived from the corresponding energy conservation equation. This equation allows to take into account the feedback between neutronic system and processes in the coolant. The thermal resistance of the gap between fuel and cladding and the effect of heat transfer through the cladding can be taken into account using the appropriate energy conservation equation for the cladding. Neglecting the heat conduction in axial direction in the fuel and cladding, these equations will be first order differential equations. According to this fact, these equations can be built into energy equation of the fluid in form of an integral kernel.
The models considered below are more or less simplified realizations of the thermohydraulic system mentioned. The complex description can be realized in the model presented in chapter 4.

In many practical cases the pressure effects can be neglected and the fluid can be considered to be incompressible. In this case the momentum and energy equations can be solved separately from each other. The state vector of the thermohydraulic system contains the fuel and coolant temperature and the coolant velocity.

The coolant temperature fluctuation in the reactor channel can be caused in this system from outside by the temperature and velocity fluctuations at the reactor inlet and by the heat generation noise in the fuel. The later is very important the forrelationship between the state of the coolant and neutron noise.

The coolant temperature fluctuations $\tilde{T}(z,\omega)$ can be written as follows:

$$\tilde{T}(z,\omega) = \tilde{T}_{in}(\omega) e^{-a(\omega)z} + \tilde{p}(\omega) F_1 e^{-a(\omega)z} I(z,\omega) - \tilde{V}(\omega) F_2 e^{-a(\omega)z} I(z,\omega) \tag{1}$$

where

$$I(z,\omega) = \int_0^z e^{a(\omega)z'} q(z')dz' \tag{2}$$

$V$ is the coolant velocity,

$z$ are the axial coordinate and the angular frequency,

$\tilde{p}(\omega)$ and $\tilde{V}(\omega)$ denote the power and velocity fluctuations, respectively,

$\tilde{T}_{in}(\omega)$ is the fluctuation of the coolant temperature at the core inlet,

$q(z)$ the axial power distribution and $q_0$ is the heating power,

$a(\omega)$ is a complex function expressing the propagation and damping due to so-called wall effect. Its form depends on the modeling of the fuel to coolant heat transfer (i.e. using effective heat transfer coefficient $h$ or taking into account the cladding). In the case of the two-temperature model $a(\omega)$ is expressed by the following equations:

$$Re(a(\omega)) = \frac{h}{V C_c} \omega \left(1 + \frac{C_f}{C_c} \frac{h^2}{(\omega C_f)^2 + h^2} \right) \tag{3}$$

$$Im(a(\omega)) = -\frac{h}{V C_c} \omega \left(1 + \frac{C_f}{C_c} \frac{h^2}{(\omega C_f)^2 + h^2} \right)$$

$C_c$ and $C_f$ are the coolant and fuel heat capacity, respectively,

$F_1$ and $F_2$ are functions of $\omega$, their form depend also on the method of modeling of the fuel to coolant heat transfer.

In the case of two-temperature model $F_1$ and $F_2$ are as follows:

$$F_1 = \frac{1}{C_c V h + i\omega C_f} \quad \text{and} \quad F_2 = \frac{q_0}{C_c V^2} \tag{4}$$

The features of the temperature noise in the reactor channel are mainly determined by the integral $I(z,\omega)$ in Eq. 2. It is obvious, that the characteristics of the temperature noise have sink structure.

We show this feature considering the APSD of the temperature noise at the core outlet i.e. at $z=H$, $H$ is the core height. In this case one can write:

$$\text{APSD}^T (H, \omega) = \text{APSD}^T_{in} (\omega) e^{-2HRe(a(\omega))} + (\text{APSD}^P(\omega) |F_1|^2 + \text{APSD}^V(\omega) |F_2|^2) \frac{B^2 S(\omega)}{|a^2 + b^2|^2} \tag{5}$$

where

$$S(\omega) = 1 + e^{-2\omega} \cos(\Im \omega)$$

$\text{APSD}^T_{in}$, $\text{APSD}^P(\omega)$ and $\text{APSD}^V(\omega)$ are the power spectra of the inlet temperature, heating power and velocity fluctuations, respectively.

Here we set for $q(z)=\sin(Bz)$, where $B$ is the geometrical buckling, $B=\pi/H$.

There are minima in the temperature noise APSD at frequencies:

$$\text{H } \Im(a) = 2k\pi \quad ; \quad k = 0,1,2,... \tag{6}$$
The values of the sink frequencies and the slope of the APSD depend on the characteristics of the heat transfer between the fuel and coolant and on the thermodynamic properties of the fuel and fluid (see e.g. Egeli et al., 1976). There are several models of the temperature fluctuation, which can be derived from the Eq.1.: 

- Simple propagation noise, i.e. if we take into account only the first term in the Eq. 1. and neglecting the wall effect. In this case one can write

\[ \text{Re}(a(\omega)) = 0; \quad \text{Im}(a(\omega)) = \omega / V \]  

(7)

This model is widely used by many authors.

- Simple propagation of inlet temperature noise taking into account the wall effect. In this case the first term of Eq. 1. is considered (present paper).

- Temperature noise in the reactor channel caused by inlet temperature and heating power fluctuations. In this case the first and second terms in Eq.1. are considered (used e.g. by Kosály and Meskó, 1972).

2.1.2. Two-phase flow, subcooled boiling

Under certain conditions the thermal boundary layer at the cladding surface can be overheated so, that is enough for the bubble formation. The vapour bubbles departure from the heated wall and travel with the fluid. After certain time they collapse because of heat transfer between bubbles and surrounding subcooled liquid bulk. There is the case of nonequilibrium two-phase flow, which can be described by conservation equation of mass, momentum and energy written for each phase. The mass, momentum and energy transfer between phases and between channel wall and each phases must be known too.

The state vector of the coolant contains the pressure, velocity and enthalpy of each phases and the void. The model should describe the whole channel, i.e. the lower part with one-phase flow and upper part with two-phase flow. The description of the thermodynamic noise in the case of nonequilibrium two-phase flow is given by Katona, 1982.

Usually simple theoretical noise models are applied in the previous works.

The simplest model can be derived for the case of saturated boiling in the whole channel. In this case the noise source is the propagating void fluctuation \( \xi(z, \omega) \) caused by random bubble generation on the heated surface (the fluctuations of other state variables of the two-phase flow are neglected). We can write for the noise source the following equation:

\[ \xi(z, \omega) = \xi(z_0, \omega) \exp \left( - \frac{i \omega}{V} z \right) \]  

(8)

where \( \xi(z_0, \omega) \) the perturbation at certain point \( z_0, z_0 = 0 \).

In the case of subcooled boiling two-phase flow the condensation of the vapour bubbles in the subcooled liquid bulk must be considered.

The equation describing the propagating void fluctuations can be derived from the mass conservation equation for the vapor phase introducing into this equation a condensation term (the fluctuations of other state variables of the two-phase flow are neglected). In this case the primary noise source is the random bubble generation on the heated surface (see Meskó and Katona, 1981). The CPSD between void fluctuation at axial points \( z_1 \) and \( z_2 \) can be written:

\[ \text{CPSD}(z_1, z_2, \omega) = \text{APSD}(z_2, \omega) \exp \left( - a(\omega)(z_1 - z_2) \right) \]  

(9)

The function \( a(\omega) \) expresses the propagation and damping due to condensation of vapour bubbles, we set for \( a(\omega) \) the following expression:

\[ a(\omega) = - \left( \frac{i \omega}{V} + \frac{\lambda_b}{V} \right) \]  

(10)

where \( \lambda_b^{-1} \) is the average bubble life time, \( (V/\lambda_b) \) is the condensation length \( l_c \).
A more sophisticated description of the void fluctuation in subcooled boiling is given by Katona and Mesko, 1981. In this model the subcooled boiling two-phase flow in the channel is described by energy conservation equation of the liquid-vapour mixture and mass conservation equation of the vapour phase. The condensation rate considered to be dependent on the liquid subcooling and the liquid temperature depend on the enthalpy of condensation of the vapour. In this case the liquid temperature fluctuates because of the temperature noise affecting on the two-phase flow from the part of the channel with one-phase flow. The liquid temperature fluctuations itself are described by Eq.1. According to this model the void fluctuations can be expressed by the following equation:

\[
\alpha(z, \omega) = \bar{T}(z_{ob}, \omega) (1 - \exp\left(-\frac{\lambda_{s} + \lambda_{f}}{V}(z-z_{ob})\right) \exp\left(-\frac{i \omega}{V}(z-z_{ob})\right))
\]

(11)

where \(\lambda_{f}^{-1}\) is the time constant of the liquid temperature feedback in the condensation process, \(z_{ob}\) the axial coordinate of onset of subcooled boiling, \(\bar{T}(z_{ob}, \omega)\) is given by Eq.1.

It is obvious that the propagating void fluctuations have the same sink structure as the temperature noise \(\bar{T}(z_{ob}, \omega)\).

2.1.3. Classification of the models of the thermohydraulic noise source

The models of the thermohydraulic noise source applied in the theoretical interpretation of the sink-structure of the neutron noise spectra can be summarised and classified as given in Table 1.

<table>
<thead>
<tr>
<th>MODELS</th>
<th>ONE-PHASE FLOW</th>
<th>TWO-PHASE FLOW</th>
</tr>
</thead>
<tbody>
<tr>
<td>1. Simple propagating perturbation</td>
<td>1.1. Inlet temperature noise, 1.2. Saturated boiling void fluctuations, Eq.8. see first term in Eq.1, and Eq.7. are valid</td>
<td></td>
</tr>
<tr>
<td>2. Propagating perturbation with damping</td>
<td>2.1. Inlet temperature noise +wall effect, see first term in Eq.1. and Eq.3.</td>
<td>2.2. Subcooled boiling void fluctuations see Eq.9 and Eq.10.</td>
</tr>
<tr>
<td>4. Subcooled boiling, 1D model of PWR channel</td>
<td>4.1. Temperature noise given by Eq.1.</td>
<td>4.2. Void fluctuations, condensation modulated by temperature noise see Eq.11.</td>
</tr>
<tr>
<td>5. Present model of the PWR channel</td>
<td></td>
<td>see chapter 4.</td>
</tr>
</tbody>
</table>

It must be emphasized, that the propagating perturbances (see rows 1. and 2. in the Table 1.) have the same mathematical description in both single and two phase flow cases (compare first term in Eq.1. with Eq.8. or Eq.9.).

2.2. Models for interpretation of the sinks in neutron noise characteristics

The existing theoretical models for the interpretation of the sink structure of the neutron noise can be ordered according to both thermohydraulic model applied and method of the description of the neutron response of the core. The models are given in Table 2. The thermohydraulic models are denoted by numbers according to the Table 1. In the table the mathematical identity of the propagating perturbances in one- and two-phase flow cases is taken into account.
Table 2. Models for the interpretation of the neutron noise sink-structure

<table>
<thead>
<tr>
<th>NEUTRONIC MODELS</th>
<th>THERMOHYDRAULIC MODELS APPLIED</th>
<th>1.</th>
<th>2.</th>
<th>3.1.</th>
<th>4.1</th>
<th>4.1-4.2</th>
</tr>
</thead>
<tbody>
<tr>
<td>point-kinetic</td>
<td>Kosály et al. ---</td>
<td>---</td>
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<td>---</td>
<td>---</td>
<td>---</td>
</tr>
<tr>
<td></td>
<td>1982</td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>point model + feedback</td>
<td>---</td>
<td>---</td>
<td>Kosály and Meskó, 1972</td>
<td>present paper</td>
<td>---</td>
<td>---</td>
</tr>
<tr>
<td>local noise</td>
<td>delay box</td>
<td>Meskó and Katona, 1981</td>
<td>---</td>
<td>---</td>
<td>---</td>
<td>---</td>
</tr>
<tr>
<td></td>
<td>Kosály, 1979</td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td></td>
<td>Katona and Meskó, 1982</td>
<td></td>
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<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>phenomenological model</td>
<td>Wach, Kosály</td>
<td>---</td>
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<tr>
<td></td>
<td>1974</td>
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<tr>
<td>phenomenological model + local-global interference</td>
<td>Kosály et al.</td>
<td>---</td>
<td>---</td>
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<td>---</td>
</tr>
<tr>
<td></td>
<td>1982</td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>1D, 1-group model</td>
<td>Valkó, Meskó</td>
<td>---</td>
<td>---</td>
<td>---</td>
<td>---</td>
<td>---</td>
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<tr>
<td></td>
<td>1977</td>
<td></td>
<td></td>
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<td></td>
<td></td>
</tr>
<tr>
<td>1D, 2-group model</td>
<td>Behringer et al., 1979</td>
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</tr>
</tbody>
</table>

2.3. Point models

The description of the neutron noise in point model requires the calculation of the reactivity fluctuations caused by both coolant and fuel temperature noise. Utilizing the one-group diffusion theory for 10 case and taking into account the self-adjointness of the flux we can write the following expression for the reactivity fluctuation:

$$\hat{\xi}(\omega) = (\mu_F F_3 + \mu_C) \left\{ I_1(\omega) T_{in}(\omega) - I_2(\omega) F_2 \bar{\nu}(\omega) \right\} +$$

$$\bar{\rho}(\omega) \left\{ (\mu_F F_3 + \mu_C) I_2(\omega) F_1 + \mu_F F_4 I_3(\omega) \right\}$$

(12)

where \(\mu_F\) and \(\mu_C\) are the reactivity coefficients of the fuel and coolant temperature, respectively,

$$F_3 = F_1 C_v$$ and $$F_4 = F_3 / h$$;

$$I_1(\omega) = N \int_0^H \frac{2}{\sin(\theta z)} e^{-a(\omega)z} d\theta z$$; \hspace{1cm} $$I_2(\omega) = N \int_0^H \frac{2}{\sin(\theta z)} e^{-a(\omega)z} \bar{\nu}(\theta z) d\theta z$$;

(13)

$$I_3(\omega) = N \int_0^H \frac{3}{\sin(\theta z)} d\theta z$$;

$$N = \int_0^H \frac{2}{\sin(\theta z)} d\theta z$$

Using the well-known relationship between reactivity fluctuations and the power noise we can write the following equation for the APSD of neutron power fluctuations:

$$\text{APSD} (\omega) = |G(\omega)|^2 \left\{ (\mu_F F_3 + \mu_C) I_1(\omega) + (\mu_F F_3 + \mu_C) I_2(\omega) \right\}^2$$

(14)

where \(G(\omega)\) is the power reactor reactivity transfer function:

$$G(\omega) = \frac{G_0(\omega)}{1 - (\mu_F F_3 + \mu_C) F_1 I_2(\omega) - \mu_F F_4 I_3(\omega) |G_0(\omega)|^2}$$

(15)

where \(G_0(\omega)\) is the zero-power reactivity function and \(P\) is the power level.
Eq. 14. expresses the generalised point model description of the neutron noise in PWR caused by fuel and coolant temperature noise due to inlet temperature and velocity fluctuations. The previous models are simplified realisations of the model above. These models can be reproduced neglecting some terms in Eq.1. or neglecting the fuel heat capacity and the feedback.

The neutron noise caused by inlet temperature fluctuations was studied by Kosály and Meskó, 1972. Their model takes into account the first term of Eq.1. and gives the neutron response to inlet temperature noise. In this case the character of the neutron noise APSD is determined by the integral $I_1(\omega)$ (see Eq.14.):

$$I_1(\omega) = \frac{2}{N^2} \frac{4 \beta^4}{|a|^2 (|a| + \beta)^2} \left( 1 + e^{-2H \text{Re}(a)} - 2 \cos(H \text{Im}(a)) e^{-H \text{Re}(a)} \right)$$

(16)

Here ,a' denotes $a(\omega)$. There are sinks in the neutron noise APSD (see Eq.14) at frequencies:

$$H \text{Im}(a(\omega)) = 2k\pi; \ k = 0,1,2,...$$

(17)

The sinks in the neutron noise APSD are mathematically "caused" by integration over the core height i.e. by the weighting of the perturbation. Obviously, the sink frequencies of the neutron noise APSD determined by Eq.17 depend on the fuel to coolant heat transfer and the ratio of the fuel and coolant heat capacity. In the case of very intense heat transfer the real part of $a(\omega)$ tends to zero. Setting for the heat capacity of the fuel $C_f < C_c$ we obtain the following expression:

$$f = k (V/H) = k/T_0; \ k = 2,3,4,...$$

(18)

where $T_0$ is the transit time through the core.

Eq.18. expresses a very simple relationship between the flow velocity and sink frequency referring to the case of propagating perturbance without wall effect.

The same result was obtained using point kinetic approximation for the description of the system response (see e.g. Kosály et al., 1982). That model can be evaluated from Eq.12. neglecting both the damping of the propagating perturbation and the feedback. In this case the CPSD (or COH) between the propagating noise at the core outlet and the neutron noise has the same sinks as the neutron noise APSD and the phase is proportional to the transit time on the half of the core length.

The values of sink frequencies depend on the weighting procedure. To show this, we consider the case of the flat flux approximation for the calculation of the reactivity perturbation (see Pdr, 1981). The neutron noise APSD contains the typical form $\sin(x)/x$. In this case Eq.18. is valid for the calculation of the sink frequencies but $k=0,1,2,...$

The neutron noise caused by velocity fluctuation was not studied in earlier works. Eq. 12. and 14. describe the sink-structure of the APSD of the neutron noise caused by both inlet temperature and velocity fluctuation too.

In this case the values of the sink frequencies differ from the values given by Eq.17. or 18. To show this, we consider the case of the velocity fluctuations only. The neutron noise spectra are determined by the form of $I_1(\omega)$.

For the sake of simplicity we use flat flux approximation. In such a way, we can obtain for the sink frequencies the following condition:

$$\min \left( 1 + \frac{1}{x^2} \sin^2(x) - \frac{1}{x} \sin(2x) \right); \ x = \frac{\omega H}{2 \pi V}$$

(19)

Finally we shall emphasize a special problem of the interpretation of the neutron noise caused by subcooled boiling. In this case the subcooled boiling (i.e. the noise source) exists usually in the upper part of the core above certain axial coordinate $z_b$. The integration in the calculation of the reactivity fluctuation will be carried out over $z_b - H$, which leads, obviously, to a very complicated picture. The relationship between the sink frequency and flow velocity could be theoretically determined if the coordinate of onset of subcooled boiling were known.
2.4. Local noise models

According to the sophisticated descriptions of the local neutron noise caused by simple propagating perturbances, there are no sinks in the local neutron noise spectra (see e.g. Behringer et al., 1979). Sinks should appear in the noise spectra, if we apply the local delay-box model for the description of the phenomenon (see e.g. Kosály, 1979). The local neutron noise can be obtained integrating the perturbation over the detector sensitive length $2l$. The integration leads to sinks of type $\sin(x)/x$. The sinks appear at the frequencies:

$$f = \frac{k/2}{T_1}; \quad k = 1, 2, 3, \ldots$$

(20)

where $T_1 = l/V$ the perturbation transit time through detector sensitive length.

The APSD and the CPSD of the neutron noise signals have the same sink-structure. The frequency values do not depend on the detector position. The CPSD (or COH) between the neutron noise and the propagating perturbation at the core outlet has the same sinks as in Eq.20, but the phase corresponds to the transit time between two measuring points.

We can observe certain minima in the CPSD between neutron noise signals caused by void fluctuations in subcooled boiling (see Meskó and Katona, 1981). In this case we consider the noise source described by Eq.9. and 10. For the calculation of the local neutron noise the well-known exponential kernel is used. The CPSD between two local neutron noise signals calculated for different values of the condensation time constant shown in Fig.1. According to this, there are periodically appearing minima in the CPSD if the average bubble life-time is short for passing the detector distance or if the condensation length $l_c$ gets comparable with the characteristic length of the local response $L$ (L is the diffusion length). These sink-frequencies depend not only on the flow velocity and detector distance but on the condensation length too, which is usually not known. It is nearly impossible to determine the flow velocity in this way.

![CPSD of the local neutron noise caused by subcooled boiling in case of different values of bubble life time $\lambda_b$.](image)

We can also predict sinks in local neutron noise spectra if we consider the noise source described by Eq.11. According to Eq.11. the frequency behaviour of the void fluctuations reflects the frequency behaviour of the temperature noise determining the bubble condensation process. These temperature fluctuations have sink character (see Eq.5. and 6.). The APSD and the CPSD of the local neutron noise caused by this noise source shall have the same sink-structure as the noise source itself.
2.5. Phenomenological approach

The model proposed by Wach and Kosály (1974) is most often used for the interpretation of the sink-structure of the neutron noise spectra. The model is based on the assumption that the neutron noise measured is a sum of a space dependent local noise (see above) and a global noise. The components of the noise are independent from each other. In mathematical sense sinks are caused in this model by addition of two independent stochastic processes, one with zero phase and the other with linear phase behaviour. There are minima in the CPSD between two in-core neutron noise signals at the frequencies:

\[ f = (2k+1)/2T_d ; \quad k=0, 1, 2, \ldots, T_d = (z_1 - z_2)/V \]  

(21)

This model itself does not predict sinks in the APSD.

Avoiding the strict theoretical restriction concerning the independency between two noise components, besides of pure global (i.e. point model) and pure local term a global-local interference term was introduced into the model (see Kosály et al., 1982). In this assumption the minima of the CPSD depend in a complicated manner on the transit time between detectors and transit time between detector position and core centre.

2.6. One-group and two-group diffusion models

The global behaviour of the core affected by propagating perturbance can be described using one dimensional one-group diffusion approximation more precisely than in point model (see e.g. Valkó and Meskó, 1977). The model derived on this basis shows, that the transfer function depends in a complicated manner on the frequency and contains periodical functions \( \sin(x) \), \( \cos(x) \), their arguments depend on the detector position or the core height. The minima appearing in the CPSD depend on the flow velocity and positions of detectors, but there isn't direct and simple relationship between the frequency of the dips and the transit times \( T_0 \) or \( T_g \). According to this theory, the point model description of the global behaviour of the core are valid in the case of small reactors and the sinks in these systems can be interpreted as in paragraph 2.3.

The two-group diffusion model gives the most exact description of the core response without feedback (see e.g. Behringer, Kosály and Pázsit, 1979). The equations describing the fluctuations of the fast and thermal flux derived from the one-dimensional one-group diffusion equations of the reactor lead to a certain eigenvalue problem. The eigenvalues of the system are generally frequency dependent and determine the core behaviour. The so called \( \lambda \) - eigenvalue, which can be approximated by \( L^2 / L \) (L is the diffusion length) determines the local response of the core. The \( \mu \) - eigenvalue determines the processes with spatial relaxation length comparable with the core size. The spectra of the neutron noise caused by propagating perturbances have been calculated for both small size and realistic energetic reactors. The APSD shows periodical dips depending on the velocity but there is no direct relationship between the frequency of minima and transit times \( T_0 \) or \( T_g \).

It is important to emphasize, that only the so called \( \mu \) component of the noise shows periodically appearing minima. It means, that the sink phenomena is connected with the response of the core with spatial scale comparable with the core length. In the case of small size core the APSD is nearly the same as in the point model and the sink frequency gives the flow velocity according to Eq. 18.

3. COMPARISON THEORY-EXPERIMENT

There is limited number of experimental works investigating the sink structure of the neutron noise in PWRs with the aim of evaluation of the thermohydraulic parameters of the core. The most important experimental results are summarised in the Table 3. For comparison two examples of saturated boiling noise are also included.
Table 3. Summary of the experimental results

<table>
<thead>
<tr>
<th>EXPERIMENT</th>
<th>NOISE SOURCE</th>
<th>RESULTS</th>
<th>INTERPRETATION</th>
<th>REMARKS</th>
</tr>
</thead>
<tbody>
<tr>
<td>Wach, Kosály, 1974</td>
<td>saturated</td>
<td>-sinks in CPSD between in-core neutron noise</td>
<td>phenomenological model, Eq.21.</td>
<td>Good agreement</td>
</tr>
<tr>
<td>Lingen, BWR</td>
<td>boiling (Eq.8.)</td>
<td>-sinks in CPSD between in-core neutron noise</td>
<td>point kinetic model, Eq. 18.</td>
<td>Global-local interference term in near-field</td>
</tr>
<tr>
<td>Kosály et al., 1982</td>
<td>air-water loop noise APSD</td>
<td>-sinks in far-field neutron noise</td>
<td>-&quot;waves&quot; in CPSD between in-core neutron signals; sinks in CPSD between TC and ex-core neutron noise; introduced into Eq.18. second sink not used /TC top of core thermocouple/</td>
<td>Small-system behaviour. No good explanation with inlet temperature noise only. The experimentally obtained constant relating 1st sink frequency with V can be explained with velocity noise; Eq.19 gives good results for 1. and 2. sinks</td>
</tr>
<tr>
<td>UWNR</td>
<td></td>
<td></td>
<td></td>
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<tr>
<td>Upadhyaya, Turkcan, 1984</td>
<td>nominal PWR conditions</td>
<td>-sinks in CPSD between in-core neutron signals; dip at 9.5 Hz in COH</td>
<td>point model approach for first sink at 2.2-2.4 Hz; empirical constant</td>
<td></td>
</tr>
<tr>
<td>Borssele, PWR</td>
<td></td>
<td></td>
<td></td>
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</tr>
<tr>
<td>Grondey et al., 1982</td>
<td>fluctuation of inlet temperature and/or boron conc. possible subcooled boiling void fluctuation</td>
<td>-sink in APSD between in-core neutron noise at 2.5 Hz; sinks in COH at 13 Hz and above</td>
<td>point model for first sink at 2.5 Hz; sinks in COH at 13 Hz and above</td>
<td>Global behaviour of the core</td>
</tr>
<tr>
<td>Obrigheim PWR</td>
<td></td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>Bastl et al., 1982; KWU PWRs 600, 850, 1300 MW fluctuations</td>
<td>-sinks in COH between in-core neutron noise</td>
<td>phenomenological model, Eq.21.</td>
<td></td>
<td></td>
</tr>
<tr>
<td>Lehman, 1984 Gössen PWR 970 MW</td>
<td>propagating noise source</td>
<td>-sinks in CPSD and COH between in-core neutron noise signal pairs</td>
<td>phenomenological model; measured values of sink frequencies are averaged Eq.21. used</td>
<td>Deviation between velocity values calculated from sink frequencies increases with decreasing flow V</td>
</tr>
</tbody>
</table>

According to the Table 3, it is not a simple task to determine which of the models reviewed above should be applied for the interpretation of the sink structure of measured noise spectra. It is possible to observe very large differences in the behaviour of similar type of reactors (compare KWU types PWRs of Borssele, Gössen and Obrigheim). This makes necessary to elaborate a more sophisticated model, which can describe the complex behaviour of the core.

4. ONE-DIMENSIONAL COUPLED THERMOHYDRAULIC-NEUTRONIC MODEL

The thermohydraulic part of the model describes both the fuel and the coolant behaviour. The equations for the small perturbations of the state variables of the coolant can be derived from the conservation equation of mass, energy and momentum -in general case- for both phase. The energy conservation equation for the fuel is built into the energy conservation equation of the coolant in form of
an integral kernel. The heat conductance inside the pellet in cylindrical coordinates, the heat resistance of the gap between fuel and cladding and the heat transfer through the cladding can be treated in analogous way in the model. This allows to take into account the feedback between neutronic and thermohydraulic systems. The thermohydraulic model of the core including the case of the nonequilibrium two-phase flow has been given by Katona (1982).

In the neutronic part of the model the two-group diffusion approximation of the transport equation is used with one group of delayed neutrons. For the description of small fluctuations around the steady-state of the state-vector the first order perturbation theory is used. For the final form of the neutronic equations obtained in this way see Kozma (1985).

In the case of one-phase flow in the reactor channel the state-vector $\textbf{N}(z,t)$ of the coupled thermohydraulic-neutronic system contains the fast and thermal neutron fluxes $N_1(z,t), N_2(z,t)$, respectively - density of precursors $c(z,t)$, $T_f(z,t)$ and $T_c(z,t)$ the fuel and coolant temperature respectively, pressure $p(z,t)$ and flow velocity $v(z,t)$.

In the one-dimensional case the thermohydraulic equations are first order partial differential equations with respect to the time and axial coordinate. For the coupling of this system with one-dimensional two-group diffusion neutronic model new variables - the first derivatives of the fluxes $\tilde{N}(z)$ and $\tilde{N}(z,t)$ with respect to the axial coordinate are introduced. The delayed neutrons are treated in the same manner as the fuel temperature. The fluctuating part of the state-vector will be:

$$\tilde{\textbf{N}}(z,t) = (\tilde{N}_1, \tilde{N}_2, \tilde{N}_1, \tilde{N}_2, \tilde{p}, \tilde{v}, \tilde{T}_c)$$

After Fourier-transforming with regard to the time the dynamic equations of the model can be written in the following matrix form:

$$\begin{align}\begin{bmatrix} U(z,\omega) \end{bmatrix} = \tilde{\textbf{N}}(z,\omega) + A'(z,\omega) \tilde{\textbf{N}}(z,\omega) = \begin{bmatrix} Q(z,\omega) \end{bmatrix}\end{align}$$

Here $\textbf{U}$ and $\textbf{A}'$ are matrices containing the physical parameters of the model, $\textbf{Q}(z,\omega)$ is the vector of driving sources distributed in the core.

The relationship describing the boundary conditions is:

$$\begin{align} \begin{bmatrix} \textbf{B} & \textbf{C} \end{bmatrix} \begin{bmatrix} \tilde{N}(z=H,\omega) \\ \tilde{N}(z=0,\omega) \end{bmatrix} = \begin{bmatrix} \phi \end{bmatrix} \end{align}$$

where $\textbf{B}$ and $\textbf{C}$ are constant matrices and $\phi(\omega)$ is the vector of inlet thermohydraulic fluctuations. We introduce the following notations:

$$\begin{align} &\textbf{F}(a,b) = \int_a^b \text{A}(s,\omega)ds \\ &\text{UKU} = \begin{bmatrix} U-1 & U-1 \end{bmatrix} \\ &\text{B}_\text{in}(\omega) = \phi(\omega) \text{B}(\omega) \\ &\text{UKU} = \int_0^z \begin{bmatrix} U-1 & U-1 \end{bmatrix} dz' \\ &z' \in [0,\infty) \end{align}$$

where $\text{UKU}$ is the variance-covariance matrix of axially distributed noise sources. Using Eq.23. the CPSD matrix between fluctuations of the state-vector at axial positions $z_1$ and $z_2$ will be calculated as follows:

$$\begin{align} &\tilde{N}(z_1,\omega) \tilde{N}^*(z_2,\omega) = \begin{bmatrix} \textbf{B} + \textbf{C}e & -\int_0^H \text{A}(z',\omega)dz' \\ -\int_0^H \text{A}(z',\omega)dz' & \text{B} + \textbf{C}e \end{bmatrix} \times \\ &\times \begin{bmatrix} \text{B}_\text{in}(\omega) & \text{F}(0,z_1) \text{B}^* + \textbf{e} & \text{F}(z_1,H) e \\ -\int_0^H \text{A}(z',\omega)dz' & -\int_0^H \text{A}(z',\omega)dz' & \text{B} + \textbf{C}e \\ 0 & -\int_0^H \text{A}(z',\omega)dz' & \text{B} + \textbf{C}e \end{bmatrix} \times \\ &\times \begin{bmatrix} \text{B}_\text{in}(\omega) & \text{F}(0,z_2) \text{B}^* + \textbf{e} & \text{F}(z_2,H) e \\ -\int_0^H \text{A}(z',\omega)dz' & -\int_0^H \text{A}(z',\omega)dz' & \text{B} + \textbf{C}e \\ 0 & -\int_0^H \text{A}(z',\omega)dz' & \text{B} + \textbf{C}e \end{bmatrix} \end{align}$$

(26)
In the calculations the eigenvalues of the system matrix
\[ A = U^{-1}A' \]  
play an important role. The eigenvalues contain all information about the physical behaviour of the system. As it is shown in earlier works (see Kozma, 1985, Kozma and Katona, 1986) the system has seven eigenvalues \( \lambda \). A pair of them slightly depend on frequency, and can be approximated as
\[ \lambda_{1,2} = \pm \lambda^{-1} \]  
The remaining 5 eigenvalues \( \lambda_3, \ldots, \lambda_7 \) are depicted on the Fig.2. as the function of the frequency (for details see Kozma, 1985).

Fig.2. The eigenvalues of the two-temperature system. Circle markings stand for frequency values: 1, 2, 3 and 4 Hz, respectively.

Neglecting the feedback and using simplified noise source our model leads to the results of two-group diffusion theory. The same is valid for the eigenvalues, i.e. in limiting case \( \lambda_{1,2} \) corresponds to the \( \lambda \) eigenvalue and \( \lambda_{3,4} \) to the \( \mu \) eigenvalue, while \( \lambda_{5-7} \) reflect the thermohydraulic properties of the system.

5. DISCUSSION OF THE RESULTS

Using the coupled thermohydraulic-neutronic model the position and the sharpness of the sinks in the characteristic functions of the noise can be calculated. It will be shown, that the earlier introduced eigenvalues of the coupled model are very useful when analyzing the sink structure of the noise spectra. Let us see first the eigenvalues corresponding to the "local-global" model.
For this reason we neglect the feedback and apply a simple propagating noise source in our model. The analysis of the model shows, that sharp sinks appear in the APSD at frequencies:

$$ f = \frac{V}{H} \frac{(2k+1)}{2} \quad k = 0, 1, 2, \ldots \quad (29) $$

In this case the \( \mathcal{A}_5 \) root writes (see Kozma, 1985):

$$ \mathcal{A}_5 = i\omega / V \quad (30) $$

That means sinks take place, when:

$$ \text{Re}(\mathcal{A}_5) \quad \text{and} \quad \text{Im}(\mathcal{A}_5) = (2k+1)(\mathcal{A}/H) \quad k = 0, 1, 2, \ldots \quad (31) $$

This condition means, that sinks appear in the spectra, if a system eigenvalue at a certain frequency coincides with one of the following points \( C_k \) of the complex plane:

$$ C_k = \{ 0, (\mathcal{A}/H)(2k+1) \} \quad k = 0, 1, 2, \ldots \quad (32) $$

Consequently we obtain sinks at the frequencies, when the curve of \( \mathcal{A}_5 \) passes through points \( C_k \).

In a more sophisticated coupled model condition expressed by Eq.32. doesn't fulfill exactly. E.g. taking the two-temperature model (see Kozma, 1985), the curve of the \( \mathcal{A}_5 \) root doesn't cross the imaginary axis and \( \text{Re}(\mathcal{A}_5) \) remains positiv in the whole frequency range (see Fig. 2.).

So in the two-temperature model - and in other sophisticated models, also the Eq.32. doesn't work. Nevertheless, at certain thermohydraulic circumstances the curve \( \mathcal{A}_5 \) goes "not far" from the points \( C_k \), i.e. in the PSDs there are still sinks. But the sinks of the coupled model are moved on the frequency axis of the system and the sharpness of the sinks decreased. This complicated picture could lead to several kinds of misunderstandings.

The present coupled thermohydraulic-neutronic model allows to treat these questions. For example, in the followings it will be shown, how the fuel to coolant heat transfer coefficient \( h \) affects the sharpness of the sinks. On the Fig. 3. the \( \mathcal{A}_5 \) curves for different \( h \) values are shown.

According to Fig. 3. with increasing \( h \) the sharpness of the sinks decreases. In this case the distance of \( \mathcal{A}_5 \) -curve and the points \( C_k \) is the largest.

Consequently in the latter case the fulfillment of condition given by Eq.32. is the worst, so the sinks must be less sharp.

For the dependence of \( \mathcal{A}_5 \) on the flow velocity i.e. for the fulfillment of the condition given by Eq.32. see Kozma and Katona, 1986.

![Fig. 3. The dependence of \( \mathcal{A}_5 \) on the fuel-to-coolant heat transfer coefficient \( h \).](image-url)
Finally, we can demonstrate the model calculating the neutron noise spectra. The Fig. 4. shows the dependence of the calculated CPSD between two in-core neutron noise signals on the flow velocity and heat transfer coefficient. The results are in accordance with the results obtained from the system eigenvalues.

![Graph showing CPSD between two in-core neutron noise signals on flow velocity and heat transfer coefficient.](image)

**Fig. 4.** Dependence of the calculated CPSD between two in-core neutron noise signals on the flow velocity, $V$ and heat transfer coefficient $h$.

The calculation of the CPSD between in-core neutron noise and temperature fluctuations at the core outlet leads to the similar consequences according to dependence of the sink structure on the parameters of the system, see Fig. 5.

![Graph showing CPSD between in-core neutron and temperature noise at the core outlet on flow velocity and heat transfer coefficient.](image)

**Fig. 5.** Dependence of the CPSD between in-core neutron and temperature noise at the core outlet on the flow velocity, $V$ and heat transfer coefficient $h$.

As it is seen from the analyses, described in this paper, the sink structure of the neutron noise characteristics in PWR's depends on the complex of thermohydraulical-neutronphysical features of the core. The values of the sink frequencies cannot be trivially related with the velocity of the coolant by the help of a simple algebraic formula.
It is not by chance, that the published measurements can be interpreted by different simple models and the velocity value can be calculated from the sink frequency with relatively great error.
The simple models cannot take in consideration the complexity of the real reactors, and as such, they can be used for some factual systems only.
By the help of the proposed model we have demonstrated the influence of the core parameters on the sink-structure of the noise spectra. According to this, the value of the sink frequency depends on the flow velocity but the relationship between the velocity and sink frequency, the frequency value and the sharpness of the sinks are influenced by the other system parameters, especially by the fuel to coolant heat transfer coefficient i.e. by the feedback between thermohydraulic and neutronic subsystems.
The calculation have shown the features of the physical processes which cause the sink-structure of the noise spectra. The evaluation of the measurements might be successfully provided using the thermohydraulic and neutronical data of the given reactors in the calculations with the model.
Finally we can conclude, that the eigenvalues of the coupled system are useful and considering the complexity of the processes - relatively simple tools in the investigation of the features of the noise spectra in PWRs.

REFERENCES

RESULTS AND INTERPRETATION OF MULTIVARIATE AUTOREGRESSIVE ANALYSIS APPLIED TO THE LOFT REACTOR PROCESS NOISE DATA

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ABSTRACT – Multivariate noise analysis of power reactor signals is useful for characterizing baseline operation and plant diagnostics, for isolating process anomaly from sensor maloperation, and for automated plant monitoring. We have developed a systematic and reliable procedure of accomplishing these tasks by improving upon the previously established techniques of empirical modeling of fluctuation signals in power reactors. The application of the algorithm to data from the Loss-of-Fluid Test (LOFT) reactor showed that earlier results (based on physical modeling) regarding the perturbation sources in a pressurized water reactor (PWR) affecting coolant temperature and neutron power fluctuations can be explained using multivariate autoregressive (MAR) analysis. This methodology has important implications regarding plant diagnostics, and system or sensor anomaly detection.

1. INTRODUCTION

The primary objective of multivariate autoregression (MAR) modeling of dynamic signals is to establish the characterization of a plant subsystem defined by a set of signals in a specified frequency range. In this light, it is necessary to choose a proper set of measurements to develop the MAR models; a poor model is often due to a lack of complete measurement set. Several authors have applied this method for diagnostic analysis of pressurized water reactors (PWRs) and boiling water reactors (BWRs) (Upadhyaya et al., 1980; Oguma, 1982; Oguma and Turkcan, 1985). Our current study applied to both power plants (Glockler and Upadhyaya, 1987) and process control systems (Upadhyaya et al., 1987) demonstrated that a systematic interpretation of the signatures derived from the MAR models can be used for plant diagnostics and for detecting and isolating process anomaly (mechanical components, controllers, etc.) and sensor maloperation.

Several issues will be addressed in this paper. The development of an appropriate MAR model requires a systematic approach to the problem as enumerated here. (a) Data acquisition, choice of sampling frequency, choice of signal set, the latter requiring some experimentation. (b) Understanding of the limitations of measurements, location of sensors in relation to the dynamic variable being measured. (c) Data qualification, choice of data length, and data compression. (d) Estimation of MAR model parameters. The model order will also depend on the purpose of the model. More comments will be made later. (e) Computation of spectral domain signatures and interpretation of their mutual relationship. (f) Derivation of diagnostic information through information management schemes.

Noise signal measurement and diagnostics is not routine in current U.S. power plants. The requirements of data acquisition facility must be given full consideration during the design phase and must be incorporated in future plants. The cost of a data acquisition system is very small compared to the total plant cost, and the benefit gained by automated monitoring and predictive maintenance will be significant.

A good collection of papers on multivariate spectrum analysis may be found in Modern Spectrum Analysis, II (Keeler, 1986). High resolution spectral analysis for short data records and multichannel spectral estimation is described by Marple (1987). Another area of interest in digital signal processing is the use of bispectrum developed via AR modeling of nonGaussian or certain nonlinear signals (Raghuvire et al., 1985) that result in higher frequency components, such as due to quadratic coupling. The bispectrum of a third-order stationary signal

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is defined by (at frequencies $f_1$ and $f_2$).

$$B(f_1,f_2) = \int_{-\infty}^{\infty} \int_{-\infty}^{\infty} R(\tau_1,\tau_2) \exp(-j2\pi(f_1\tau_1+f_2\tau_2))d\tau_1d\tau_2$$

(1)

where $R(\tau_1,\tau_2)$ is the triple correlation of the signal.

The theory of MAR modeling and improvements in model selection procedure are described in Sect. 2. Section 3 contains a full description of the derivation of spectral signatures, spectral decomposition, and some new interpretation results. We will also make some comments on bispectrum analysis to detect frequency coupling. The methodology developed in this paper is applied in Sect. 4 to operational data from the LOFT reactor primary system. Summary and concluding remarks are given in Sect. 5.

2. MULTIVARIATE DATA-DRIVEN MODELING OF PLANT SIGNALS

Our primary goal is in establishing a characterization of the interrelationship among a set of signals from a plant subsystem. Implicit in this approach is the assumption that the fluctuating (or the dynamic) components of the signals represent also the physical dynamics of the system. If this is not true, then the developed models represent nonphysical relationship among the signals, and will be useful for representing the characteristics of a network of sensors. The multivariate autoregressive (MAR) models used in this study are estimated directly from the data, and we will refer to them as data-driven models. The models are adaptive to the data and can easily be transformed to the frequency domain to determine signatures used in diagnostics studies.

2.1 Multivariate Model Structure

Let $\mathbf{X}(t) = \{x_1(t), x_2(t), \ldots, x_n(t)\}$ be a vector of jointly wide-sense stationary processes. The causal relationship (including feedback) among them may be modeled by

$$\mathbf{X}(t) = \sum_{i=1}^{n} A_i \mathbf{X}(t-i\Delta t) + \mathbf{V}(t)$$

(2)

where $\Delta t$ is the sampling interval (sec), and $\{A_1, A_2, \ldots, A_n\}$ are $(m \times m)$ parameter matrices. $\mathbf{V}(t)$ is a vector of external driving noise sources of the equivalent multivariate linear system. Without loss of generality, $\mathbf{V}(t)$ is considered as a white noise sequence, with mean zero and covariance matrix

$$\Sigma = \text{E} \{\mathbf{V}(t)\mathbf{V}(t)^T\}.$$  

(3)

The "order" of the model $n$ depends on the bandwidth of the signals, sampling interval $\Delta t$, and the resolution required in the frequency domain.

2.2 Model Parameter Estimation

The estimation of parameters $\{A_1, A_2, \ldots, A_n\}$ makes the assumption that the parameters are time-invariant for the data-block used in the modeling. We have used a recursive approach for the computation of $\{A_1, A_2, \ldots, A_n\}$. At any stage $n$, the parameters are computed by updating the parameters of the $(n-1)$ order model. Equation (2) is written in the correlation domain using the Yule-Walker formulation as

$$C(k) = \sum_{i=1}^{n} A_i C(k-i), \quad k=1,2,\ldots,n$$

(4)

where $C(k)$ is $(m \times m)$ correlation matrix at lag $k$.

$$C(k) = \text{E} \{\mathbf{X}(t)\mathbf{X}(t-k\Delta t)^T\}, \quad C(-k)=C(k)^T$$

(5)

The recursion developed by Levinson and Durbin (Whittle, 1963; Durbin, 1960) uses the symmetric block-Toeplitz property of the correlation matrix.
Multivariate autoregressive analysis

\[
C = \begin{bmatrix}
C(0) & C(-1) & \ldots & C(-n+1) \\
C(1) & C(0) & C(-1) & \ldots & C(-n+2) \\
& \ddots & \ddots & \ddots & \ddots \\
& & C(n-1) & C(n-2) & \ldots & C(0)
\end{bmatrix}
\]  

(6)

The correlation functions are calculated using the biased estimate

\[
C_{ij}(k) = \frac{1}{N} \sum_{t=k+1}^{N} x_i(t)x_j(t-k)
\]

(7)

The parameter sets are calculated recursively using the Levinson-Durbin algorithm (see Upadhyaya et al., 1980). The noise covariance is given by

\[
\Sigma = C(0) - \sum_{i=1}^{n} A_i C(i)^T
\]

(8)

2.3 Comments on Model Order, \( n \)

In a theoretical sense, the model order \( n \) is selected such that the prediction error is minimized. This approach is not very useful in practical applications. Excessive high order for the model will result in unstable and inaccurate estimation, because the changes in the correlation functions stabilize at large lags. Moreover, a very high order may not make physical sense, for at these levels we will be trying to model measurement noise in addition to the process noise. It is important to realize that we want to extract the dynamics of the system, rather than the dynamics of the entire measurement. Thus some experimentation is necessary for a defined set of signals from a given subsystem to arrive at an optimal model order. Akaike (1974) derived a measure based on maximum likelihood approach. Called the Akaike Information Criterion (AIC), the order \( n \) is selected such that the AIC is minimized.

\[
AIC(n) = \text{N} \ln(\text{det}(\Sigma)) + 2n^2N, \quad N = \text{number of data samples.}
\]

(9)

Further study by Kashyap (1980) showed that AIC is not a consistent measure and often overestimates the order as the data length increases. A modification to AIC was suggested by Rissanen (1986), termed the minimum description length (MDL).

\[
\text{MDL}(n) = \text{N} \ln(\text{det}(\Sigma)) + n^2 \ln(N).
\]

(10)

We want to state that the optimality of the model is also influenced by the combination of signals, and some of the spectral domain signatures reflect the overall adequacy of the model.

When signals contain measurement noise, the MAR model resolution will be degraded, because an all-pole model will be converted to a pole-zero model. It is necessary to remove the effect of this noise (Kay, 1980). One procedure is to shift the correlation functions, thus avoiding the use of \( C(0) \).

3. SYSTEM CHARACTERIZATION IN THE FREQUENCY DOMAIN

Once the optimal MAR model is established, the model is transformed to the frequency domain, and several diagnostic-related signatures are calculated for the individual signals and for signal pairs. These are,

(a) auto- and cross power spectral density functions (APSD and CPSD),
(b) ordinary coherence (COH) and partial coherence (PCOH), and corresponding phase functions,
(c) ordinary and partial contribution functions to a given signal APSD (NSCR and PNSCR) from various noise sources,
(d) transfer functions among the measured signals,
(e) inherent noise source spectra, coherence and phase functions.

Our intention is to combine the various signatures systematically for diagnostics purposes.
3.1 Determination of Spectral Signatures

The (mxm) power spectral matrix is derived directly from the model and is given by

$$S_{xx}(f) = H(f)^{-1} \sum H^*(f)^{-1} \Delta t, \quad |f| \leq \frac{1}{2\Delta t}$$  \hspace{1cm} (11)

where the multivariate transformation matrix is

$$H(f) = I - \sum_{k=1}^{n} A_k \exp(-j2\pi fk\Delta t)$$  \hspace{1cm} (12)

(* indicates complex conjugate transpose). The diagonal elements $S_{ii}(f)$ are the APSDs and the off-diagonal elements $S_{ij}(f)$ are the CPSDs. The contribution to the APSD of signal $x_i$ from driving noise source $v_j$ is defined by

$$NSCR_{ij}(f) = \frac{|\{H(f)^{-1}\}_{ij}|^2}{S_{ii}(f)} \sigma_{jj} \Delta t$$  \hspace{1cm} (13)

The APSD $S_{ii}(f)$ contains contributions from all transfer function paths and thus $SCR_{ij}$ may not always represent direct effects between two signals, but includes all possible effects.

Remarks: When $\sum_{j=1}^{m} SCR_{ij} + 1$, indicates the goodness of the data-driven model, proper signal combination and model order for the defined frequency range.

3.2 Spectral Decomposition to Study Direct Effects

The true relationship among a set of process signals in terms of signal-to-signal effect may be obtained by decomposing the frequency domain transformations (Oguma, 1982; Oguma and Turkcan, 1985). The equations are written in the form

$$X(f) = G(f) X(f) + W(f)$$  \hspace{1cm} (14)

where

$$G_{ij}(f) = -\frac{H_{ij}(f)}{H_{ii}(f)} \quad \text{and} \quad G_{ii}(f) = 0$$  \hspace{1cm} (15)

The inherent noise power spectral matrix of $W(f)$ is defined by ($W_{ij}(f) = V_{ij}(f)/H_{ij}(f)$)

$$Q_{ij}(f) = E(W_{ij}(f)W_{ij}^*(f)) = \frac{\sigma_{ij}}{H_{ij}(f)H_{jj}^*(f)}$$  \hspace{1cm} (16)

When the inherent noise PSD $Q_{ij}(f)$ is equal to the signal PSD $S_{ij}(f)$, the signal $x_i$ may be treated as one of the independent driving sources in the system. When this property is satisfied, this will be a very important aspect of the diagnostic interpretation. In general the driving noise sources at different frequencies are attributed to different signals. The NSCR and FNSCR should be considered as the effect of one signal over another signal, represented in this form mathematically through associated transfer function dynamics. When none of the inherent noise PSDs coincides with the corresponding signal PSD, this indicates a deficiency in the signal set and the choice of signals must be reconsidered.

The partial effect of signal $x_i$ on signal $x_j$ is expressed by the "peeled" components in the form

$$\begin{bmatrix} 1 & -G_{ij} \\ -G_{ji} & 1 \end{bmatrix} \begin{bmatrix} x_i(f) \\ x_j(f) \end{bmatrix} = \begin{bmatrix} W_i(f) \\ W_j(f) \end{bmatrix}$$  \hspace{1cm} (17)

Equation (17) is then used to represent the partial effect of one signal over the other by computing partial coherence and partial signal contribution ratios. The partial noise source
contribution ratio from signal \( x_j \) to signal \( x_i \) has the form

\[
PNSCR_{ij}(f) = \frac{\text{NUM}_{ij}(f)}{Q_{ii}(f) + |G_{ij}(f)|^2 Q_{jj}(f) + 2\text{Re}[G_{ij}(f)Q_{ij}(f)]}
\]

where

\[
\text{NUM}_{ij}(f) = \begin{cases} 
Q_{ii}(f) & \text{if } i=j \\
|G_{ij}(f)|^2 Q_{jj}(f) & \text{if } i \neq j
\end{cases}
\]

These quantities relate two signals \( x_i \) and \( x_j \) directly by excluding the effects of other sources. The spectral descriptors outlined above may be used to generate a signal transmission path diagram (STPD), relating the diagnostic cause-and-effect mechanisms. Choice of an incomplete signal set, or unmeasurable process variables that influence the dynamic behavior, will cause distortion in the STPD and can provide improper diagnostic information. Our experience has also shown that a logical management of the spectral information will provide the detection and isolation of process anomaly (mechanical components, controllers, etc.) and sensor maloperation.

4. APPLICATION TO DIAGNOSTIC ANALYSIS OF THE LOFT REACTOR

The above methodology was applied to noise signals from the LOFT reactor at operating power levels of 25%, 50%, 75% and 100%. LOFT is a 55 MWth PWR containing extensive in-core instrumentation. The major objectives of our study are

(a) to determine the primary source of perturbation causing fluctuations in in-core neutron detector and core-exit thermocouple signals,

(b) to evaluate the added advantage of using reactor coolant pump differential pressure and core differential pressure for primary system diagnostics.

4.1. Data Description

The data acquisition was performed by the Instrumentation and Controls Division of Oak Ridge National Laboratory. We performed the digitization at the University of Tennessee as follows:

- Anti-alias filter setting = 4 Hz.
- High-pass filter setting = 0.05 Hz.
- Sampling frequency = 10 Hz.
- Total data length = 1500 sec.

The following signals are used for various runs.

- In-core neutron detector signals (NE-5DB-11,17,44,63).
- Core-exit thermocouple (TC) signals (TE-5UP-3, TE-2UP-5).
- Differential pressure transducer (ΔP) across the primary coolant pump (PDE-PC-1).
- ΔP across reactor core (PDE-PC-6).
- ΔP across the steam generator primary side (PDE-PC-2).

4.2. Results of Analysis in the Frequency Range 0–5 Hz

The combination of signals pump ΔP(PDE-PC-1), core ΔP(PDE-PC-6), in-core neutron noise (NE-5DB-61) and core-exit TC(TE-5UP-3) is used to develop a multivariate AR model. The order of this model is estimated to be \( n=40 \). First a summary of the results of this analysis is given. This will be followed by some conclusions (diagnostic results).

In each of the following cases, no feedback from the second signal to the first was observed in the frequency range 0–5 Hz (that is, \( NSCR_{12}(f)=PNSCR_{12}(f)=0 \)). The partial coherence \( PCOH_{12}(f) \) and the partial signal contribution from the first to the second signal \( PNSCR_{21}(f) \) were identical. The degree of signal coupling was measured using ordinary and partial coherence functions, and signal contribution ratio functions. For all the signal pairs considered here, the phase angle is linear as a function of frequency (indicating the presence of propagating effect).

(a) The APSD functions of the four signals (solid line) and those of the inherent noise source or residual noise components (dashed line) are shown in Fig. 1. Any difference between the two APSD alterations indicates the effect from other signals.
(b) Comparison of pump Δp and core Δp signals (Fig. 2). Approximately 80% of the core Δp fluctuations in the frequency range 0-4 Hz originate from the pump Δp perturbations. The coherence between the two inherent noise sources is zero, indicating their independency.

(c) Comparison of pump Δp and in-core neutron flux signals (Fig. 3). The COH and NSCR21 functions are identical and are slightly greater than the corresponding partial functions. The direct transmission path from pump Δp signal to in-core ND signal is significant. The phase plot is linear up to 4 Hz, and the phase shift at low frequency goes to zero. The inherent noise components of the two signals are again independent of each other.

(d) Comparison of noise signals pump Δp and core-exit TC (Fig. 4). The COH and NSCR21 functions are identical, and the phase is linear up to 2 Hz starting from -180 deg at zero frequency. The corresponding PCOH and FNCR21 have low values indicating an indirect signal path from pump Δp to core-exit temperature.

(e) Comparison of signals core Δp and in-core ND signals (Fig. 5). High coherence but low partial coherence is observed in the range 0-4 Hz. Also low values for NSCR21 and FNCR21 are seen. The direct relationship between core Δp and core-exit TC signals is again very small, thus indicating that the pump Δp is the primary perturbation source.

(f) Comparison of in-core ND and core-exit TC signals (Fig. 6). High coherence but smaller partial coherence. But the PCOH and PSCR21 are higher than for the relationship between pump Δp and core-exit TC. The phase plot is linear in the 0-2 Hz region, and tends to -180 deg at zero frequency.

4.3 Diagnostic Interpretation of Results

The following are the results of our analysis based on the comparison of the above frequency domain signatures.

1. The sum of the signal contribution ratio functions for each of the above four signals is plotted in Fig. 7. The sum of NSCR functions is unity, indicating the validity of the model fit. This is a significant result of our analysis, suggesting further that the combination of signals is appropriate.

2. The perturbation source in the LOFT reactor primary system is the pump Δp fluctuation and has a frequency range of 0-4 Hz. This variable influences all the other measured signals through propagating processes. This result was also derived by Shih et al. (1987).

3. There is a direct signal path from pump Δp fluctuation to in-core neutron flux fluctuation. This is evidenced by high coherence (ordinary as well as partial) and a linear phase behavior between the signals.

4. The ordinary coherence between pump Δp and core-exit TC signals is high with linear phase, but the partial effects are small. This indicates that there is only a small direct effect between these two signals. The core-exit temperature fluctuation is caused by neutron flux fluctuation.

5. The calculated transit time between the in-core ND signal and the core-exit TC signal depends only on the axial position of the core-exit thermocouple. LOFT reactor is essentially a point reactor.

The complete cause-and-effect relationships are described by the signal transmission path diagram (STPD) of Fig. 8. This representation is very useful for establishing the baseline system behavior and for routine noise surveillance. The sum of the estimated delay times along two different paths coincide, indicating another characteristic of this system.

5. SUMMARY AND CONCLUDING REMARKS

The results of application of multivariate data-driven modeling methodology and a systematic interpretation of the frequency domain signatures to the LOFT reactor process noise demonstrate the feasibility of applying such techniques to commercial nuclear power plants. By monitoring the various diagnostic signatures it is possible to detect and isolate sensor malfunction and process anomalies (mechanical components, controllers, etc.). Signal acquisition and analysis should be part of routine plant monitoring (Valko et al., 1985). The current work in progress includes:

(a) Developing an automated procedure for identifying signal transmission path networks and establishing a database of STPDs for different operational conditions of a commercial power plant.
Multivariate autoregressive analysis

(b) Application of process control systems, such as the Aluminum Company of America's Hot Line Rolling facility in Alcoa, Tennessee.
(c) Development of a pattern matching technique to isolate system and sensor anomalies.

Most of the digital signal processing (DSP) algorithms perform repetitive calculations. One example of this type of calculation is the estimation of correlation functions. Special DSP chips are now available in the market that "exploit the repetitive nature of signal processing by pipelining the data flow for extra speed," (Aliphas and Feldman, 1987). It is necessary to interface the DSP software with DSP chips to perform certain calculations, thus increasing the speed and precision of arithmetic operations.

We are also looking into bispectrum analysis technique to determine the relationship between a pair of frequency components of the same signal. This may be developed using triple correlations and generalized autoregressive models and applied to systems exhibiting quadratic coupling, nonharmonic and nonlinear effects.

ACKNOWLEDGMENTS

The research reported here is supported in part by the U. S. Department of Energy, Office of Technology Support Program. We extend our thanks to the Instrumentation and Controls Division, ORNL, for providing us with LOFT reactor operational data.

REFERENCES

Fig. 1. Power spectral density functions of the signals (solid line) and those of the inherent noise components (dashed line). (a) $\Delta p$ across the primary coolant pump (PDE-PC-6), (b) $\Delta p$ across the reactor core (PDE-PC-1), (c) in-core neutron detector (NE-5DB-61), and (d) core-exit thermocouple (TE-5UP-3). LOFT reactor data at 100% power and flow.

Fig. 2. (a) Ordinary and (b) partial coherence, and the corresponding phase functions between pump $\Delta p$ and core $\Delta p$ signals.

Fig. 3. (a) Ordinary and (b) partial coherence, and the corresponding phase functions between pump $\Delta p$ and incore neutron detector signals.
Fig. 4. (a) Ordinary and (b) partial coherence, and the corresponding phase functions between pump $\Delta p$ and core-exit thermocouple signals.

Fig. 5. (a) Ordinary and (b) partial coherence, and the corresponding phase functions between core $\Delta p$ and core-exit thermocouple signals.

Fig. 6. (a) Ordinary and (b) partial coherence, and the corresponding phase function between incore neutron detector and core-exit thermocouple signals.
Fig. 7. Sum of the normalized contribution functions to different signals from the individual noise sources, (a) pump $\Delta p$, (b) core $\Delta p$, (c) in-core neutron flux and (d) core-exit coolant temperature.

Fig. 8. Signal transmission path diagram (STPD) identified from the multivariate modeling of the LOFT reactor core subsystem in the frequency range 0-5 Hz.
OPERATIONAL EXPERIENCE (PART II)

Session chairman: E. Türkcan (Netherlands)
SUMMARY OF THE SESSION

In the second part of the operational experience session, five papers were presented, all giving the operational experience of the different country's NPP experiences.

- The first paper is presented by Turi et al., who discuss the primary circuit vibration noise of the VVER-440 FWR's. Two twin-units of PAK's FWRs each with 6 primary system loops. Complete vibration test including pressure and neutron noise is given.

- The second paper is presented by Michel and Puyl. This gives very systematically large amount of operational information on the French FWRs' noise experiences. The view is extended to 56 different power plants worldwide. LPM and VM system capabilities has been reported for all French Reactors. The paper gave results of an extensive cost/benefit analysis made by EDF for the French reactors, with the benefit over 30 years experience. It seems that the benefit/cost ratio obtained is about two to one.

- The third paper is presented by Fry who summarized the 4th Informal Workshop Meeting in the USA. The paper contains the views of many well known specialists and the experiences of utilities in Germany and France, together with the Finnish view. It seems that:
- LPM and noise diagnostics are alive and well established in the USA,
- Application in US plants will increase as noise analysis is shown to be useful in day-to-day plant operation,
- Informal annual workshops provide a means for plant engineers to share experiences in use of noise analysis for plant diagnosis.

- The fourth paper is presented by Eklund et al., giving the view of early fault detection and diagnosis in Imatra Voima's Oy's Power Plants. Computerized condition monitoring of Lovisa Nuclear Power Station (2x455 MWe) was discussed and also vibration analysis and diagnosis was given for present and future monitoring systems. Model-based failure diagnosis and knowledge-based approach to condition monitoring are outlined. This paper also gives
- evaluation of residual lifetime of components,
- online process models for failure diagnosis,
- conditioning monitoring System NATALI.

- The fifth paper by Schutte et al. explained the utility experience in reactor noise analysis in the German LWR's Biblis (Schutte), Obrigheim (Sommer) and GKN (Weingarten). The GRS evaluation and experience are also compiled in this paper.
- GRS studies on guidelines for LPM and VM for early failure detection are summarized.
- Know-how transfer to the utility to support noise measurements and preparation of data bank for noise analysis are summarized and several utility experiences with details were nicely presented.

In this session we learned about the operational experiences from five countries. The high benefit of the noise analysis over the cost is very promising for the future. In due course, advances in electronics and microprocessors, including special signal processing capabilities for early failure detection and monitoring systems, with integrating expertise in the systems, will become useful for the utilities.
PRACTICAL EXPERIENCE WITH PRIMARY CIRCUIT VIBRATION NOISE ANALYSIS AT PAKS NUCLEAR POWER PLANT

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Automatic vibration monitoring systems have been installed for units 1-4 of the PAKS Nuclear Power Plant. During normal operation regular routine measurements are carried out on the units to analyze the changes in the vibration and to forecast the failures. The results of the measurements have been stored for later comparisons and analysis. In parallel with these tests the vibration measurements carried out during course of start up of the unit, have the aim to study the complex movement of the primary circuit, to determine the different forms of movements and to analyse the vibrational behavior of the units at different operational conditions. In relation with these tasks some special vibration measurements had taken place.

In this paper an overview of normal measurements and some results are given. In connection with the evaluation of the measurements of start up, some eigenfrequencies and modes, representing the global movements of the main equipments were determined. Our efforts concentrate to examine the changes of these vibration characteristics related to the condition of the unit and the parameters of the primary circuit.

1. INTRODUCTION

The noise diagnostic system of the Paks Nuclear Power Plant has been operating since starting the operation of the first unit in 1983. We have already introduced the diagnostic system of the first two VVER-440 PWR twin units at the previous SMORN conferences (1,2). The system installed with the 3\&4 units has essentially the same conception as the system of units 1 and 2, but significantly surpasses the previous ones in respect of the construction and services.
The positions of the accelerometers and pressure transducers arranged on the 3rd unit is shown on figure 1. Each main equipment is supplied with one accelerometer of vertical direction, with the exception of the main closing valve in the cold leg; according to our experience, due to the closeness of the main circulating pump it does not provide additional information as compared to the data measured on the main coolant pump. Pressure measurements are provided after the main circulating pumps, and at the outlet and inlet points of the core. There are 4 accelerometers on the stud bolts of the reactor and another 6 sensors on the protecting tubes of the control and safety assemblies on the upper block.

Acoustic emission sensors are used for monitoring the leakage of in-core detectors at the outlets on the top of the vessel and at the safety valves on the pressurizer.

For measuring the reactivity fluctuations the signals of ionization chambers are used. The number of the in-core measuring chains is varying in the units, signals of 3 channels on the 1st unit, 6 channels on the 2nd unit, and 16 channels on the 3rd and 4th units (7 pcs of SPND per channel) can be used; in addition the fluctuation signals of some thermocouples can also be processed.

We expect from the noise diagnostic measurements that the development of the failures can be prevented by the early detection of the defects. Noise diagnostic tests are performed in almost every nuclear power plant, but the frequency and scope of the test are varying. In addition to the regular, and continuous monitoring one can find measurements performed per cycles and occasionally. The problem is, that one can not determine in advance which defects can be detected through the noise diagnostic measurements. In case of phenomena which are well-known in their well-defined cause and effect relations the checking process can be expressed in algorithm and the warning and alarm levels can be defined. In most cases, however, unexpected phenomena occur and only the condition being different from the normal one can be detected. For investigation of the abnormality we want to set up a spectrum library. In a power station the background of vibration continuously changes due to the different operational conditions, therefore instead of investigation of the effect of different technological situation on vibration spectra, we determine the eigenfrequencies of main components of primary circuits by means of calculations and experiments, and sensitivity tests are carried out, that is effects of changes of strain, temperature, pressure, load are determined by calculation and experimentally if possible.
Our activity can be classified into three main groups:
1. Routine measurements for monitoring the well-known phenomena which have been proved with cause and effect relations.
2. Extension of the data library, for studying the vibration behaviour of the primary circuit and development of measuring methods.
3. Investigation of the reasons of abnormal vibrations.

2. REGULAR MEASUREMENTS UNDER NORMAL OPERATION OF THE UNITS

The frequency of the tests to be performed at Paks Nuclear Power Plant is determined on the basis of the expected time of the development of the defect. Table 1. provides a review of the regular repeated measurements and of their frequency.

2.1 Monitoring of vibration of main coolant pumps

According to our experience the development of the defects detectable by vibration diagnostics is slow, checking is performed once a day at present. Defects, developing more rapidly, can be detected by standard plant instrumentation. The daily checking measurement includes spectrum records up to 500 Hz from which the vibration amplitudes at the rotation frequency, \( f_0 \), and the frequency \( 2f_0 \), as well as \( 5f_0 \), \( 10f_0 \) and \( 15f_0 \) of frequencies corresponding to the blade frequencies are selected. The program displays if any of the amplitudes exceed the adjusted lower or upper limits. These limits were determined on the basis of data gathered for several years¹. The reduced data are stored. In case of exceeding the limit, a detailed spectrum analysis follows but sometimes in-site measurements are also to be performed. Time signals recorded on tape at several measuring points simultaneously provide possibility for detailed analysis. Figure 2. shows the change of the vibration amplitude in time measured on the double rotation frequency of a main coolant pump. The defects of the clutch unit with inside and outside toothing connecting the motor and the pump occur at this frequency. The usual vibration amplitude is below 0.1 ms\(^{-2}\). The defect has developed
slowly, in about 2 months, then became stationary. After shutdown of the pump and lubrication of the clutch the vibration continuously decreased, then after spreading out of the grease, the vibration gradually increased again. According to economical calculations in case of such failures it is not worth stopping the operation of the unit for repairing the pump. Although the toothing of the clutch gets damaged and needs replacement but the loss of the production would be a greater expense. Lubrication can be performed in isolated leg causing minor power loss.

2.2 Analysis of reactivity noises

Checking of the vibration of internals of the reactor vessel is performed every month. The change in the distribution of the probability density function within the range of 0.1-10 Hz of the time signals of the ionization chambers, indicates sensitively the growth of core barrel vibration. Time signals are also recorded and stored for detailed later analyses if necessary. Peak at 1.1 Hz, characteristic for the vibration of regulating rods occurs in the in-core signals. By its growing it was possible to localise a strongly vibrating control rod, which caused serious wearing on the connecting part of the head and seizure on the outer surface of the rod. Computer-aided process for localization of the vibrating rod is applied on the 3rd and 4th units. (3).

2.3 Leak detection

For checking the leakage of the primary circuit too many sensors would be required due to the 6 loops. Therefore the measurements are made only at two points: at the safety valves of the pressurizer and at the top of the vessel to check the outlet leakage of the in-core detectors. Accelerometers are applied on the safety valves of the 2nd unit. On the 3rd and 4th units AE detectors are used, RMS is produced and DC signal is transmitted to avoid the noise of the long cables.

3. TESTING THE VIBRATION BEHAVIOUR OF THE PRIMARY CIRCUIT

In respect of vibration the primary circuit consists of large masses of main equipments and elastic pipelines interconnecting them. Two methods can be used for determination of parameters describing the eigenfrequency movements of the coupled mechanical system (modal characteristics):

- finite element strength calculation for determination of the eigenfrequencies and forms of movements (calculational modal analysis), and
- the analysis of the response signal given to an outer excitement of the equipments (experimental modal analysis).

For the calculations the SAP-IV and the SAP 86 codes are used. The primary circuit is modelled by one loop, consisting of pipe elements, rods, joints and
springs and more detailed models serve for the steam generator and the reactor. For the latter one we have also developed a simple model made of coupled rigid bodies. These models are under development and experimental checking of the calculated values is carried out in parallel.

Experimental determination of the eigenfrequencies has been carried out on the basis of the results of the excitement tests on the one hand and of the analysis of the vibration signals recorded at different operating conditions and during normal operation of the units on the other hand. Later on we shall show some experimental results.

3.1. Shaker tests

![Diagram of shaker tests](image)

Prior to the hot run of the 3rd and 4th units of the Paks Nuclear Power Plant excitement experiments were performed in the primary circuit at ambient temperature in order to determine the forms of eigenfrequency movements of the main components and the loop. The exciting force was given by the electromagnetic vibrating head type Brüel and Kjær 4808 the frequency of which was swept at a rate of 0.1 Hz/s. The positions of transducers (13 pcs of Brüel and Kjær vibration accelerometers used for the measurements at the 3rd unit and the excitation points (G1,G3) are illustrated in figure 3. The exciting force was 80-100 N in different measurements, the covered frequency range was 5-80 Hz. The dynamics of the response signals can be judged from figure 4. Even the background noise (1) (shaker switched off) is rather structured at different measuring points, which is the dynamic effect of the so called micro-excitement (e.g. seismic effects, other operating units). These micromovements can be reliably measured also with accelerometers, the coherence and the phase functions between the signals of the sensors show the basic vibration modes of the system. Figure 4. illustrates that the response signals of the excitement (2) include

![Graph of response signals](image)
the frequency peaks of the micro-vibration but with a better dynamics of minimum 20 dB in spite of the low excitation force. Further tests have shown that the eigenfrequencies due to the micro-movements and the related deformed shape show good compliance with the values determined by the shaker test. Interpretation of the results is limited only by the fact that the frequency composition of the micro-excitation is unknown. As an example for the results of the excitation tests, figure 5 shows the frequency transfer function of the acceleration signal measured at the 5th point (horizontal motion of the steam generator) to the G1 excitement. Based on the figures we can mark three resonance peaks (7.85 Hz, 11.83 Hz, 13.35 Hz). On the basis of the Nyquist-diagram one can determine the exact value of the resonance points and the parameters of the simplified model system having one degree of freedom. In the range of 5-80 Hz we could identify 9 eigenfrequencies of the system consisting of the steam generator - main circulating pump - pipeline. We have to emphasize however, that the test were performed at special conditions of the unit (~30°C, ~3 bar in the primary circuit) and the main and auxiliary systems of the primary circuit did not operate. Further test should be carried out in order to analyze the change of these eigenfrequencies due to the effect of the temperature and to study their presence in the signals of the sensors used during normal operation.

Fig. 5. Transfer function between signal at point 5 and excitement at G1

Fig. 6. Vibration signals of SG (3G1) and MCP (3F1)

1. SG vertical motion
2. SG longitudinal motion
3. SG transversal motion
4. Motion of hot leg (horizontal)
5. Harmonics of complex loop motion
6. MCP pendulum motion
7. MCP vertical motion
when the signal to noise ratio is known. From the results of the above test we obtained some characteristic eigenfrequencies and related forms of movements which could be identified in the spectra registered during the start up of the 3rd unit. As an example fig.6. shows the vibration signal of the steam generator and that of the main circulating pump. One can see that the motion of the main equipments of the primary circuit is similar to the coupled mechanical systems and the eigenfrequencies can be detected all over the loop due to the tight connection among the components.

3.2 Special measurements during start up and normal operation of units

The vibration signals measured on head of the reactor vessel (R1-R4) of the operating units show similarity in their nature. According to the coherence analysis eigenfrequencies can be identified in the frequency ranges 10-18 Hz, 25-30 Hz, 50-55 Hz, 103-110 Hz, 130-135 Hz, 140-148 Hz. These eigenfrequencies have similar values for each unit. As we have already mentioned the coherence usually is zero and the phase is fluctuating due to the low signal to noise ratio of the accelerometers below 10 Hz. Test were performed during the start up of the reactor in order to identify the forms of motion. The effect of change in the temperature of cooling water and of metal structures is shown in relation with the measurements of the 2nd unit (figure 7.)

According to the analysis systematic change of frequency can be observed in the ranges of 25-30 Hz, 50-55 Hz, and 130-150 Hz which is proportional to the change of the elasticity modulus (E) with the temperature; consequently they correspond to global vibration. A mathematical model made up of coupled rigid bodies has been developed to analyze the motion of the reactor vessel. The model describes the vertical and pendular motions of the multiple mass pendulum system. According to the calculations the 8 eigenfrequencies belonging to the four mass points (vessel, pit, core barrel, block of protecting tubes) are as follows: 3.4 Hz, 4.1 Hz, 5.8 Hz, 13.1 Hz, 17.5 Hz, 26.5 Hz, 53.6 Hz, 130.5 Hz. The eigenfrequencies over 10 Hz show good agreement with the measurements. The 53.6 Hz and the 130.5 Hz are related to the vertical movement and
the 26.5 Hz to the pendular motion and their values are affected, first of all, by the stiffness of the spring connecting the reactor vessel and the pit, as mass points. As the temperature changed variation in the elasticity of this torus spring could have been recognised. The transducers measuring the pressure fluctuation signals are connected to the sampling points by pulse tubes. Due to technical reasons the length of pulse tubes is about 30-40 m at the 1st unit and 2-4 m at the other units. From the evaluation of the results, the pressure fluctuation signals reflect mainly the response of these tubes of different lengths, and the flow noise and pressure fluctuation due to the movement of the equipments are superimposed on it.

As for the pressure signals measured in different cooling loops the autospectra show strong similarity in nature, but the coherence is high only at 0.66 Hz, which can be attributed to the change of the water level in the pressurizer. The other tests also show that apart from this common effect, the coherence measured between the pressure signals of the different loops is low. Figure 8. shows the pressure signals measured in the 2nd loop of the 2nd unit. From their change the following can be concluded:

- The 7.1, 21.3, 42.6 and 98 Hz peaks (marked with f1, f2, f3, f5) are displaced to 9.1, 27.3, 54.5 and 124 Hz. The frequency shifts reflect the change of the actual sound speed in the cooling water. Consequently the 7.1 Hz and its upper harmonics (21.3, 42.6 Hz) and the 98 Hz are standing waves of the primary circuit.

- In the frequency range marked with f4 one can observe a different behaviour. The frequency shift is proportional to the change of the sound speed with the pressure under constant temperature (~ 50 °C) and consequently this peak can be attributed to the eigenmotion of the pulse tube.

The coherence between the pressure and vibration signals, suggests that the pressure components interpreted as standing waves of the primary circuit mean forced excitation for the main equipments, while feedback effect of the vibrations of the mechanical equipments can not be identified in the pressure signals.
The behaviour in the higher frequency ranges can be further illustrated with the results obtained in course of cooling down of the units. Figure 9. shows the coherence function of the F1 pressure signal and the F1 vibration signal of the 2nd unit in the range of 50-150 Hz. It can be seen that the pressure fluctuation caused by the standing wave in the primary circuit means forced excitement for the main cooling pump, having vibration frequency component proportional to the change of the sound speed.

For studying the eigenfrequency movement of the loop we have analyzed the instationary process of the varying forced excitement in course of the run-out of the main circulating pump. Figure 10 illustrates the results obtained during a planned shut down of the 3rd unit. In the vertical movement of the main circulating pump both the decreasing 25 Hz and the 50 Hz components emphasize the 18 Hz component, which is the eigenfrequency of the vertical movement. Due to the uncertainty the evalution (the phenomenon is changing in course of the observation time, too) the parameters determined from the measurement can be considered only as a rough estimation.

Fig.9. Coherence between vibration of MCP and pressure signals at different temperature and pressure

Fig.10. Excitation of eigen-frequencies during run-out of MCP
4. STUDYING THE REASONS OF ABNORMAL VIBRATIONS

The main function of the automatic vibration monitoring system is to monitor the changes which occurred in the measured data and to display them. However, any diagnostic activity is of adaptive nature, that is after analysing the abnormality the experiences can be built into the monitor functions. After detection of the abnormality we perform on-site, multi-channel measurements (for the sake of reproducibility usually in fixed measuring positions) for offline evaluation.

The above remarks once again show that one of the main areas of the interpretation activity is the development of special software for evaluation and its application to automatic monitoring the system.

References


OPERATIONAL AND ECONOMICAL EXPERIENCE WITH VIBRATION AND LOOSE PARTS MONITORING SYSTEMS ON PRIMARY CIRCUITS OF PWRs

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Abstract - A world survey made by EDF in 1986 has shown that many utilities are concerned with vibration and acoustic monitoring systems, especially for loose parts detection, reactor internals monitoring and pumps surveillance. Systems installed in all PWR's of Electricité de France, since 1970 for the Chooz prototype reactor and since 1976 for the 900 MW reactors, are operated on a regular basis. They are able to detect significant incidents which are recalled.

In this paper, costs of equipment, development, maintenance and operation are compared to the benefits gained from the systems. The main benefit is an early detection of potential damages, which results in saving repairs expenses and plant outage time.

In conclusion, the benefit to cost ratio is highly positive.

1. INTRODUCTION
EDF now operates 45 PWR reactors of both 900 MW and 1300 MW types, the average operating time being 5.2 years and the first reactor having 10 years of operation. Reactor noise monitoring systems were installed on all reactors and operated since the first startup of the plants.

In 1986, it seemed it would be useful to take stock of the situation and gather available information on the systems operating experience. A joined committee of three EDF Departments - Nuclear and Fossil Generation, Plant Design and Building, Research and Development - worked on the benefits of the reactor noise monitoring systems as far as plant operation and availability were concerned. A survey was also made to know the operating experience with these systems on reactors abroad and see how it compares with EDF experience.

2. WORLD EXPERIENCE WITH REACTOR NOISE MONITORING SYSTEMS
The survey was made through the INPO and NOMIS networks and data was collected for 56 reactors (37 in the United States, 9 in Federal Republic of Germany, 3 in Sweden, 3 in Switzerland, 2 in Belgium, 2 in Finland).

The results are as follows:
- All 56 reactors have a loose parts monitoring system continuously operated. The system uses accelerometers in most cases (52 reactors) and acoustic detectors on 4 reactors.
- Data analysis is accomplished by plant staff (22 reactors) or by the system supplier specialists (17 reactors) or by plant staff with the help of specialists for detailed analysis (17 reactors).
- 28 reactors (50 %) have a vibration monitoring system of the reactor internals. The system is continuously operated on 20 reactors and periodically operated on 8 reactors. The system uses accelerometers (16 reactors) expansion meters (3 reactors) seismic displacement sensors (3 reactors) neutron noise measurements (6 reactors).
Data analysis is accomplished by plant staff (14 reactors) or by the system supplier specialists (4 reactors) or by plant staff with the help of specialists for detailed analysis (10 reactors).
- A few incidents are reported:
  - loose parts in the steam generator inlet
  - loose bolts of the lower part of reactor internals
  - failure of hold down springs of upper core structure (detected by the vibration monitoring system)
  - the neutron noise monitoring system identified a situation in which the internals were "cocked". The next refueling outage, when the internals were removed, a bolt was found underneath one edge and had caused the internals to be cocked about 3/8 inch.

3. EDF OPERATING EXPERIENCE WITH REACTOR NOISE MONITORING SYSTEMS

3.1. Systems description

All EDF reactors are equipped with a loose parts monitoring system and a vibration monitoring system of the internals. The systems were designed by EDF Research and Development Department.

Loose parts monitoring is achieved by accelerometers, vibration monitoring uses both accelerometers and ex-core neutron detectors.

Loose parts monitoring is continuously operated with alarms in case of impacts detection. Vibration monitoring is periodically operated and detailed analysis of the vibration spectra is carried out. Both systems are operated by plant staff but detailed analysis of the routine vibration recordings as well as analysis of incidental situations are accomplished by EDF Research and Development Department.

The vibration monitoring system for the 1300 MW reactor will be continuously operated and will detect abnormal phenomena.

3.2. Monitoring systems costs

Costs of loose parts and vibration monitoring systems are the summation of the following items:
- design and development
- equipment
- systems operation.
- Design and development costs.

These are the expenses of EDF Research and Development Department for the initial design and development of the systems, their later improvements, plant personnel training and all analyses of routine vibration recordings and incidents.
- Equipment costs.

As systems were regularly improved, equipment slightly changed with the different types of reactors.
- Operating costs.

Loose parts monitoring does not require a significant involvement of plant personnel except in case of impact detection and vibration monitoring mainly requires routine recordings.

3.3. Incidents detection and systems benefits

Incidents detected until now on EDF PWR's involved the loose part monitoring system. This system was able to detect loose parts and was also used to monitor the vibrations of incore instrumentation thimbles on 1300 MW reactors by noise measurements.

Vibration monitoring systems were able to measure abnormal vibrations such as fuel rod vibrations due to baffle jetting and could differentiate the vibration levels of different types of fuel elements but did not detect any incident of consequence for plant availability.

Incidents relevant to loose parts and thimbles vibration are recalled in the next chapters and benefits gained from the loose parts monitoring systems are estimated in terms of plant availability.

3.3.1. Loose parts incidents

BUGEY 5: during the preoperational tests of the reactor in October 1978, a provisory instrumentation had been set inside the inlet box of a steam generator. This instrumentation got loose as soon as the reactor coolant pumps were operating, impacts were detected by the monitoring system and the pumps were stopped in a very short time. There was no significant damage to the steam generator.

Two years later the same incident occurred on the SAINT LAURENT B1 reactor but the monitoring system was not operating and damages were noticed later on. Repair works had to be done on the steam generator and plant start-up was delayed by about 2 weeks.
In conclusion, early detection on BUGEY 5 saved 2 weeks of plant unavailability plus repair costs.

FESSENHEIM 1: the nut of a guide tube pin was detected inside the inlet box of a steam generator in March 1982 but the reactor coolant pumps were kept operating for 4 days and the steam generator was damaged. It took about 2 weeks to do the repair works. The damages to the inlet box and the tubes of the steam generator were important enough to get a good estimate of the consequences of any similar loose part. It can be conservatively assumed that the damage would have not been significantly more severe if the pumps had been in operation for more than 4 days, so that the benefit of the monitoring system cannot be taken into account.

BUGEY 2: in July 1982, a loose part of a guide tube pin was detected inside the inlet box of a steam generator but this time the reactor coolant pumps were quickly stopped. The prompt reaction of the plant operators saved repair costs and outage time.

The gains of the monitoring system are assumed to be equal to the losses of the FESSENHEIM 1 incident, i.e. 2 weeks of plant unavailability plus steam generator repair costs.

FESSENHEIM 1: a hook used during maintenance activities was left inside the inlet box of a steam generator and impacts were detected a few days after plant start-up in September 1985. It was possible to make an analysis of the frequency and energy of the impacts and compare the noise signature of the loose part with the 1982 incident on the same reactor. According to this analysis, the potential damage due to this new loose part would be quite lower than in 1982 and the decision was to continue reactor operation for the whole fuel cycle. At the end of the cycle there was no significant damage to the steam generator.

Although the loose part had been detected by the monitoring system, its size and weight were such that it was of no consequence for the steam generator. If there had not been any monitoring system, reactor operation would have continued in the same way without any equipment damage.

In this case again, one has to conservatively assume that the monitoring system was of no benefit.

TRICASTIN 4: in March 1987 impacts were detected inside the inlet box of a steam generator as the reactor was leaving cold shut down conditions. Reactor coolant pumps were stopped within a few hours. A nut of guide tube pin was found inside the steam generator. There was some damage to the inlet box but repair was not necessary.

The monitoring system saved repair costs and 2 weeks of plant unavailability as it did for BUGEY 2 in 1982.

3.3.2. Incore instrumentation thimbles

After a few months of operation of the first 1300 MW units, incore instrumentation thimbles were worn out because of flow-induced vibrations and several leaks occurred.

The loose part monitoring system proved to be very helpful as accelerometers mounted on the thimbles guide tubes at the bottom of the reactor vessel could detect thimbles stricking their guide columns in the reactor internals.

For the 1300 MW reactors, the system was used to:
- make the difference between several modifications of the reactor internals (spring caps, sleeves,..) whose purpose was to reduce thimbles vibrations.
- There was a full scale test on the FLAMANVILLE 1 reactor.
- Check the efficiency of the modifications on the 1300 MW reactors.

If the loose part monitoring system had not been available, thimbles wear would have had to be measured using eddy current techniques. This meant:
- to operate the reactor for a period of time long enough to get measurable wear
- to take the reactor to cold shutdown conditions
- to measure thimbles wear using eddy currents
- to take the reactor back to full power conditions.

Each thimbles noise monitoring made with the accelerometers saved at least taking the reactor to zero power and back to full power plus the cost of wear measurement using eddy currents. As thimbles noise monitoring was carried out 18 times on 1300 MW reactors, it is easy to point out the benefit gained from the loose part monitoring system.

3.3.3. Costs and benefits balance

Reactor noise monitoring systems costs are quite well known and increased as the number of reactors in operation increased in the past recent years.

Benefits gained from the loose part monitoring system are calculated with the following assumptions:
- early detection of the loose parts for BUGEY 5 in 1978, BUGEY 2 in 1982 and TRICASTIN 4 in 1987 saved the expenses of the repair works on the steam generator, based on what was spent for FESSENHEIM 1 in 1982 and SAINT LAURENT B1 in 1980,
- early detection also saved two weeks of plant unavailability whose cost was quite high in 1978.
At the beginning of the nuclear program in France, fossil fired plants were used for energy production when a nuclear reactor was not in operating conditions and the price of fuel or coal was important. As more and more nuclear plants were connected to the electric network, any unavailable reactor could be replaced by other reactors output and the unavailability cost decreased.

- Thimble noise monitoring on 1300 MW reactors saved plant unavailability in 1985 an 1986 as well as expenses relevant to thimbles wear measurements using eddy currents.

Fig. 1 shows the cumulative costs and estimated benefits and Fig. 2 shows the costs and benefits balance. All numbers are in million of 1986 french francs (MFF).

Fig. 1: cumulative costs and benefits
Fig. 2: costs and benefits balance

4. CONCLUSION

EDF commitment for reactor noise monitoring systems has been of great interest until now. Loose parts monitoring systems, in particular, helped reduce potential damages to the steam generators and plant unavailability. The cumulative total of expenses and estimated benefits for 45 PWR's since 1976 (900 MW and 1300 MW reactors) shows a benefit – cost ratio of about two to one.

REFERENCES


SUMMARY OF THE FOURTH CONFERENCE ON
UNITED STATES UTILITY EXPERIENCE IN
REACTOR NOISE ANALYSIS*

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Abstract - The fourth informal conference on United States utility experience in reactor noise analysis and loose-part monitoring was held at the Northeast Utilities Service Company offices in Hartford, Connecticut, May 12-14, 1987. Host and general chairman for the meeting was J. V. Persio of Northeast Utilities. This conference provided a forum where utilities could share information on reactor noise analysis on an informal basis. There were about 60 attendees at the meeting representing 10 utilities, 3 reactor vendors, 8 consulting organizations, and 4 universities and research laboratories.

Twenty-three papers were presented at the conference, dealing with various aspects of loose-part monitoring, neutron noise analysis, and standards activities.

SESSION I: REACTOR NOISE ANALYSIS SYSTEM CALIBRATION

R. Wood of Duke Power Company described the detection of loose parts caused by failure of a main coolant pump at Oconee-3. Loose-part sensor signals were used to track loose pump parts as the primary flow carried them from the pump to the bottom of the reactor vessel. Sensors located on the control rod drives (CRD) did not detect the parts because there was no low-impedance sound path from the sensors to the vessel. Wood stated that the CRD-mounted sensors on Catawba Units 1 and 2 have been relocated to the pressure vessel lifting lugs to improve detection of loose parts. Wood went on to describe Duke Power's program of loose-part monitoring system (LPMS) calibration at its PWRs, including potential pitfalls in system calibration, and suggested solutions to some calibration problems frequently encountered.

C. Mayo of Science Applications International Corp. (SAIC) summarized his research on methods for calibrating LPMS using known masses (pendulum swing and ball drop methods). His results showed a strong relationship—as predicted by Hertz and Lamb theory—between observed vibration frequency and the time during which a loose part is in contact with the impacted structure. He pointed out that system calibrations should be performed for a range of expected loose part masses and velocities. Other factors that must be accounted for in calibration are the angle of impact, curvature at the point of impact, and if testing is performed under water, the drag imposed by water. Also, calibrations should not be performed at locations that have major internal or external structural attachments at branch points within several feet of the test area.

J. Keller of Bolt, Beranek, and Newman, Inc., described and demonstrated LPMS calibrations using various force hammers and a multichannel analyzer. He showed that force hammers impart a flat energy spectrum out to ~5 kHz or greater.

depending on hammer weight and the material used on the hammer head—the smaller the hammer and the harder the tip material, the broader the excitation spectrum. By using a load cell on the hammer and 2 spare LPMS signal channels, it was possible to record and analyze the input impact signal simultaneously with the responses from a number of LPMS sensor channels.

SESSION II: REACTOR NOISE ANALYSIS

D. Wach of Gesellschaft für Reaktorsicherheit (GRS) described the loose-part and vibration monitoring programs used in the Federal Republic of Germany. He pointed out that their LPM systems follow industry standards, which provide guidelines for calibration, sensor location, and mounting (magnetic mounts are being replaced by threaded studs because of the variable sensitivity associated with magnetic mounts). Loose parts are located by a combination of peak amplitude and time-of-arrival analysis. Examples of loose-part diagnosis included a 43-g tool found in a PWR steam generator and a BWR valve with too much clearance. Diagnosis of loose parts is based on pattern matching to the burst shapes of known noise, and on arrival time. GRS is installing LPMSs that employ both digital and analog recording techniques, with analysis performed off-site at the GRS laboratory.

Vibration and noise analysis standards provide guidance for vibration monitoring of vessel internals using signals obtained from vibration sensors placed on reactor upper head and pumps, from pressure sensors, and from neutron detectors. Measurements are made during pre-op tests, during plant startup and shutdown, and during normal operation. Extensive vibration modeling of the primary system is performed to guide interpretation and diagnosis of abnormal vibration. Examples of successful diagnosis included relaxing the tension of hold-down springs on the upper head, contacting of a main coolant pump with concrete building supports, and decreasing prestressing of secondary support screws. A current R&D goal is to identify pattern recognition discriminants that can be used for automated screening of vibration signatures. Pattern recognition discriminants have been used to monitor main coolant pumps to predict main shaft failure. Wach stated that noise monitoring has resulted in obvious cost benefits including increased plant availability, reduced radiation exposure to maintenance personnel, and improved licensing support.

F. Sweeney of Oak Ridge National Laboratory (ORNL) presented a comparison of U.S. noise monitoring programs with those of the French and Germans. Major differences exist in areas of plant personnel training, maintenance of signature libraries, and commitment of plant personnel to the best possible job of noise diagnosis. Sweeney concluded that U.S. utilities must adopt an attitude toward noise monitoring similar to that of their foreign counterparts in order to improve technical justification for plant life extension.

J. Stevens of Combustion Engineering reported a successful diagnosis of a main coolant pump problem at Millstone Unit 2 using a combination of pump vibration monitoring and core neutron noise analysis. Oil whirl in the bearings excited a pump structural resonance at 5.2 Hz, which, in turn, produced primary coolant hydraulic pulses that excited a beam mode vibration of the core support barrel. The vibration and neutron noise returned to normal after replacement of an upper bearing in the pump.

J. Robinson of Technology for Energy Corp. described the use of loose-part monitoring, time trace, frequency analysis, and neutron noise signal analysis to diagnose a flow anomaly at the Callaway PWR plant. Small (-1%) changes in power and corresponding changes in flow (-0.3%) and coolant temperature were hypothesized to be caused by a flow instability at the core inlet. This example of neutron noise analysis combined with LPM illustrates how these methods can be used to answer licensing questions without plant shutdown.

G. Zwingelstein of Electricité de France (EDF) described the state of the art and practice of noise analysis in French PWRs. They have an extensive database accumulated over 100 fuel cycles from 34 plants. EDF has used accelerometers mounted on in-core guide tubes and in-core neutron noise to diagnose and monitor abnormal vibration of thimble tubes inside the guide tubes. They now have 50 accelerometers installed at one plant (on each of the guide tubes) to obtain data for estimating the relationship between the number of impacts and wear of the tube walls. They are also using noise analysis to monitor primary pumps for
cracks in main shafts and for instrument surveillance. In one case, the lack of noise on a pressure sensor signal indicated a plugged impulse line.

Session III: LOOSE PART MONITORING

J. Phillips of Babcock & Wilcox (B&W) described an automated digital LPMS that overcomes the basic problem of systems using analog tape recorders to capture data for loose-part diagnosis. The digital system uses a circular buffer that freezes when an alarm occurs. This scheme provides immediate access to the signals that initiated the alarm. Measured time delays and signal energy are used to determine the location and significance of the loose part, thus allowing plant personnel to make a timely and careful analysis before a decision is made regarding continued plant operation. Current analog systems, on the other hand, often require extensive off-line analysis of rather sketchy data before the nature of a loose part can be diagnosed.

S. Glass described techniques used by B&W to minimize the number of false alarms in LPMS. Some of the techniques used are (1) setting the alarm threshold 3 to 10 times background level using automatic gain control techniques, (2) using high-pass filters to eliminate low-frequency noise caused by plant equipment, (3) disregarding as electrical noise any signals that show a <0.5-ms delay between channels, (4) disregarding as insignificant signals that do not repeat within 2 minutes, (5) requiring at least two channels to indicate the presence of a loose part, (6) defeating alarms that are generated by such normal occurrences as control rod drive action, and (7) using software algorithms to perform waveform discrimination.

J. Persio of Northeast Utilities summarized some of the physical characteristics of a loose part as related to the acoustic signals it generates when impacting. For example, physical characteristics such as size, shape, and material were related to signal characteristics such as time envelope, amplitude, and frequency content. His experience has shown that significant loose parts can generate signals having amplitudes of 1 to 100 "g" and frequencies of 1 to 10 KHz. He also noted that stable low-frequency plant noise, such as the blade passing frequency of the main coolant pumps, can be used to validate LPMS operation.

G. Zwingelstein summarized Electricité de France's 16 years' experience with loose-part monitoring from a cost/benefit standpoint. He estimates that the LPMS program has saved his utility ~$200 million versus a cost of only ~$28 million. The success of this program has led to development of an artificial-intelligence-based diagnostic work station that aids plant personnel in early detection of loose parts as well as reactor coolant pump and turbine problems. The work station also provides a data link to Zwingelstein's R&D group, which provides assistance in plant diagnostics and continues development of improved diagnostic methods.

O. Glöckler of the University of Tennessee described a multivariate noise analysis method useful for plant diagnostics, for isolating process and sensor anomalies, and for automating plant monitoring. The method was used to determine the cause-and-effect relationship among process signals at the Loss-of-Fluid-Test (LOFT) reactor. He concluded that this technique lends itself to automation using an artificial-intelligence-based expert system. Initial implementation of the method is planned for the Paks nuclear power plant in Hungary.

R. Kryter of ORNL introduced a novel diagnostic process based on motor current signature analysis to monitor the condition and operational readiness of electric-motor-driven mechanical equipment. The electric motor itself acts as a transducer, sensing both large and small, long-term and rapid mechanical load variations, and converting them to variations in the current induced in the motor windings. Thus, diagnostics can be performed on motor-operated valves and other similar plant equipment, without installing new sensors or wiring, using measurements made at the motor control center with portable equipment.

J. Thie proposed the use of noise analysis to extend the time between routine plant equipment checks while ensuring that instrumentation faults are detected in a timely manner. He presented several examples in which continuous noise analysis of process signals could have provided early warning of problems such as partially blocked sensing lines and failed neutron flux sensors.
R. Lyon of Massachusetts Institute of Technology presented a novel waveform analysis based on cepstrum methods, designed to eliminate the effects of multiple signal paths in acoustic and vibration signals. The method has been automated using adaptive techniques and may prove as useful in loose-part monitoring as it has in analysis of valve problems in internal combustion engines.

SESSION IV: OVERVIEW OF INDUSTRY R&D AND STANDARDS ACTIVITIES

B. Lubin of Combustion Engineering described a comprehensive program to provide five plant owners with a state-of-the-art understanding of internals vibration and loose-part monitoring. The program provides guidelines for periodic acquisition, reduction, and evaluation of data related to operation and performance of reactor systems and components. The program covers three phases of a comprehensive monitoring program: guidelines for data acquisition, reduction and diagnostics; plant-specific evaluation of current procedures and equipment compared to guidelines; and training of plant personnel in proper monitoring and diagnostic methods.

C. Mayo of SAIC presented a summary of his comprehensive research on loose-part monitoring, which was sponsored by the Electric Power Research Institute. He concluded that the combination of Hertz and Plate Wave theory can predict signal properties rather accurately and that most of the diagnostic information lies in a frequency range of 1 to 10 kHz. He recommends using good sensor mounts and a force instrumented impact hammer on a range of known masses to calibrate the monitoring system, in addition to three sensors surrounding each reactor vessel natural collection area for loose parts. His results also show that the mass of a loose part (which is required to estimate the energy imparted by the colliding part) can be inferred from the time-domain signal.

G. Zigler of Science & Engineering Associates, Inc. summarized the nuclear reactor standards activities of the ASME Subcommittee on Vibration Monitoring, including those related to piping, core support, heat exchangers, loose parts, and rotating equipment. He also described the ASME procedures for standards development and approval. He provided a status summary of the ASME Standard OM-12, "LWR Loose Part Monitoring and Diagnostics."

F. Sweeney of ORNL stated that a revised ASME Standard OM-5, "Requirements for In-Service Monitoring of Core Support Axial Preload in PWRs," will be issued by September 1987. He identified areas where significant revisions were made, including verification of core support barrel (CSB) natural vibration frequency via models or pre-op tests, requirements for more baseline data (including cross-power spectral densities), an expanded explanation of statistical errors, and cautionary statements regarding the use of scale factors to obtain quantitative estimates of CSB displacement.

In a separate presentation, Sweeney described an ORNL research program sponsored by NRC whose objective is to evaluate methods for monitoring the continued health of reactor internals during extended plant life. He stressed the importance of participation by utilities and plant manufacturers to ensure maximum benefits and usefulness to the industry. In a companion presentation Sweeney's colleague, B. Damiano of ORNL, made a plea for information regarding the mechanical details of reactor internals as input for his modeling efforts aimed at predicting the vibrational modes of LWR internal components.

CONCLUDING SESSION

The meeting concluded with general discussion regarding next year's meeting as well as ways to stimulate broader application of noise analysis in U.S. nuclear plant operations. This author suggested that noise monitoring and diagnostics will be used more widely if they can be shown to be of benefit for day-to-day routine plant operation, not just for detecting gross anomalies.

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EARLY FAULT DETECTION AND DIAGNOSIS IN FINNISH NUCLEAR POWER PLANTS

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Abstract - In Finland nuclear power plants account for one third of the production of electric energy. Finland’s four nuclear units have proved successful in terms of both environment and economy. Condition monitoring has an important role in improving safety and availability and in reducing maintenance costs. This paper describes the condition monitoring systems and methods used in Finnish nuclear power plants. Operational experiences and future planning are also reviewed.

1. NUCLEAR POWER IN FINLAND

Nuclear energy in Finland is generated by nuclear power plants in Loviisa and Olkiluoto. The companies producing nuclear energy are Imatran Voima Oy (Imatra Power Company, IVO), and Teollisuuden Voima Oy (Industrial Power Company Ltd., TVO). In recent years about 35% of electricity in Finland has been produced by nuclear power.

IVO operates two PWR units in Loviisa on the southern coast of Finland. The units are identical, having VVER-440 type reactors with six horizontal steam generators and a turbogenerator plant with two 235 MW turbogenerators.

The main components of the plant were delivered by the Soviet organisation Atomenergoexport (AEE), while Finnish suppliers delivered several pieces of auxiliary equipment and also the primary circulation pumps. Loviisa 1 started its commercial operation in 1977 and Loviisa 2 in 1980. The load factors of both units have been remarkably high, averaging 87.6% during the last five years, despite a twelve-month fuel cycle.
TVO operates two identical 710 MWe BWR units on Olkiluoto Island on the west coast of Finland. The units were delivered by ASEA-ATOM on a 'turnkey' basis. TVO 1 started its operation in 1978 and TVO 2 in 1980. The load factors of both units have been high in international comparisons.

2. COMPUTERIZED CONDITION MONITORING OF NPP LOVIISA

2.1 System description

The on-line surveillance system of NPP Loviisa has operated since 1982. Besides four turbines, vibration signals from the primary coolant pumps, control-rod mechanisms and loose-parts monitoring system are connected to the system. It is also used in reactor noise analysis. The operational experiences are good. Turbine bearing problems have been detected by the system and it has had a major role in trouble-shooting primary circulation pump sealing problems.

A principal diagram of the surveillance system is presented in Figure 1. The system is based on a PDP-11/35 computer with data-acquisition hardware. Measuring data is transferred to VAX-computers in the power plant and in the IVO Central Laboratory in Helsinki. IVO's nationwide DECnet computer network is used in the data transfer. This approach makes it possible to use the PDP solely for measurement purposes and for temporary data storage. Data evaluation and reporting are performed through VAX-computers.

![Diagram of the on-line surveillance system of NPP Loviisa](image)

Fig. 1. The on-line surveillance system of NPP Loviisa.
A 256-channel analog multiplexer concentrates measurement signals to the two-channel A/D-converter of the PDP-computer. At the present, over 200 signals are wired to the multiplexer. Most of the signals are vibration signals but also a large number of process signals is connected. The vibration signals are wired from the monitoring units of turbogenerators, primary circulation pumps and from the loose parts monitoring system. The system also includes accelerometers on primary circulation pumps and control-rod drives which are used only in computerized monitoring. The process signals, i.e. temperature, pressure, flow and neutron flux, are connected from the conventional plant instrumentation system. The surveillance system is described in more detail by Vuorenmaa et al. (1983).

2.2 Vibration analysis and diagnosis

The data transferred from the on-line surveillance system is used for long-term trend monitoring. Wide band rms-values, running speed related amplitudes, phases and static shaft positions are included in trend reports produced by programs in VAX-computers (Villanen, 1987).

In addition to long-term trend monitoring, further analyses are performed periodically as follows:

- rundown analysis,
- analysis while changing the load and
- detailed frequency and phase analysis of the bearing and shaft vibrations three times annually.

These measurements are performed with portable equipment. VAX-computers are again used for data reduction, reporting and long-term storage. The diagnostic personnel can use the information stored in the condition monitoring database and study the data with various methods. The data can be presented in many different graphical formats. These include e.g. waterfall presentation of vibration spectra and shaft vibration orbits. As an example Figure 2 shows a waterfall presentation of shaft vibration spectra of a primary circulation pump during a two-week period.
2.3 A case study of vibration monitoring

The shock-pulse monitoring of the primary circulation pump motor bearing gave an indication of a bearing problem. A bearing vibration analysis was performed with portable equipment and the results showed some indications of damages. It was decided to continue the operation of the pump and to perform extended vibration monitoring to be able to follow the development of the damage.

The computer-based trend monitoring of the shaft vibration showed no increase when the first indications of the damage were detected. This is natural, because shaft vibration is relatively unsensitive to rolling-element bearing damages. In addition, the shaft-vibration probes are placed on the pump structure—not on the motor. The trend monitoring results in Figure 3 show a slight increase during the three-month operation before the pump was repaired during the annual revision.
Fig. 3. Trend monitoring of primary circulation pump shaft vibration levels during a 180-day period.

The condition monitoring instrumentation of the primary circulation pumps was later improved with permanent accelerometer installations on the motor and pump structures.

2.4 Reactor noise analysis

Diagnostic reactor noise measurements were started at Loviisa 1 in 1978. Initially the measurements were carried out mainly with portable equipment twice during each fuel cycle. Later the surveillance system has been used for on-line noise analysis. Special measurements, e.g. consisting of many simultaneous signals, are still carried out with portable equipment.

Ex-core neutron noise and pressure noise are monitored regularly. Ionisation chambers are connected to the on-line surveillance system through isolation amplifiers and frequency-voltage converters. Dynamic pressure signals are measured from the inlet and outlet of the core, from each of the six primary loops and from the pressurizer. Rms-values of neutron signals are calculated once a month and compared to baseline values. Spectrum analysis of neutron and pressure signals is performed twice annually.

The noise patterns have been quite stable during the operational life of the power station. The core barrel motion cannot be detected under normal operating conditions. The noise analysis results are reviewed in more detail by Jokinen et al. (1986).
3 DEVELOPMENT OF CONDITION MONITORING SYSTEMS
AND METHODS IN IVO

3.1 Next-generation condition monitoring system

IVO has developed a MicroVAX-based condition monitoring system with
simultaneous multichannel data-acquisition and a digital signal pro-
cessor. The first unit has been installed to a conventional power
plant in Naantali, and the first commercial delivery has also been
started. The Loviisa diagnostic system will also be upgraded with this
new equipment in the near future.

The more powerful measuring and computing capabilities of the
MicroVAX computer make it possible to use, in addition to conventional
vibration monitoring, several new monitoring methods. Among the new
methods under testing in this modern system are the evaluation of the
residual lifetime of components and process failure diagnosis using
process models.

The monitoring of aging or residual lifetime of major structures,
which are subject to pressure is based on calculation models. In moni-
toring attention is paid on creep and/or low-frequency fatigue depend-
ing on the wall thickness. The monitoring is directed at the critical
sections of steam piping and at the turbine. This information is es-

tential in justifying plant-life extension.

Equally important with the development of new monitoring methods
have been the improvements on the user interface. By using a modern
graphics work station it is possible to create a highly interactive
user interface including windows and mouse-sensitive items.

3.2 Model-based failure diagnosis

Traditional process monitoring is based on limit and trend checking
of process variables which should stay within prescribed limits. Var-
dious failures can then be detected but the operators are alerted
only after the measuring values have changed considerably. Failure
prediction and localization or detection of beginning degradation of
components are not supported by traditional monitoring systems.

A modern approach to process monitoring is the use of process
models. A model is a mathematical description of the process. The
model can be used either to predict the operation of the process or to
estimate some non-measurable properties of the process.

A principal diagram of process monitoring by the model reference
method is shown in Figure 4. The process is modelled with a matched
filter which gives an estimate of the process output. The signal which
indicates the failure is the low-pass filtered difference between the
actual output and its estimate.
The model reference method has been used to monitor the operation of the turbine control systems and water-level control systems of the steam generators of NPP Loviisa (Eklund, 1984 and Kuusmanen et al., 1987). In the modern system of the Naantali power plant it is applied to the monitoring of coal supply systems, combustion and different components like fans, pumps and valves (Lautala et al., 1987).

3.3 Knowledge-based approach to condition monitoring

IVO has launched a research program to build knowledge-based condition monitoring and diagnostics systems. The first applications are vibration monitoring of rotating machinery and failure diagnosis using process models and multivariate signal analysis (Eklund et al., 1987). The first prototypes are scheduled for completion late in 1988.

The most important goal of the research program is to create a general tool for process modelling and control strategy of diagnosis using expert system techniques. We believe that this tool can be used in different applications, such as turbine vibration or efficiency monitoring.

A demonstration prototype of an expert system performing troubleshoot diagnosis has been developed by using OPSS production system language. The system analyses the processes, e.g. combustion and steam production, of coal-fired power plants. The localisation of the sources for disturbances is based on noise contributions produced by multivariate AR-models (Mäkkänen, 1987). The research prototype of this system will be implemented on a Symbolics 3620 machine which uses the 'Knowledge Engineering Environment' (KEE) development tool package.
4 CONDITION MONITORING AND DIAGNOSIS
IN TVO’S PLANTS

4.1 Process computer system

TVO’s both units were from the beginning furnished with general-purpose process computer systems for core surveillance, process surveillance, logging and reporting and event recording. The system is also utilised for the control-rod maneuvering function in an open-loop control mode. The man/machine interface is based on semigraphic colour crt:s and fast matrix printers in the central control room. The hardware has been completely renewed during 1986 in order to ensure capacity for a functional development of the system. As a part of the renewal core surveillance system has been realized by means of a double computer configuration to improve the redundancy in the system. The in-core fuel management system has also been updated to the COREMASTER system delivered by ASEA-ATOM.

4.2 Measuring computer system

As a complement to the general-purpose process computer system there is also a dedicated computer system for fast data acquisition, generally called the 'Measuring Computer System'. This system is utilized for disturbance recording (like the black box of an aircraft), noise surveillance and analysis and special measurements during steady-state operation or plant unit tests. The system is equipped with graphic crt:s and hardcopy devices. The basic software of the system executes data acquisition, storage and disturbance recording. Software for analysis and output in both time and frequency domain is also included. This software can be run from any work station on site. This option makes it a useful tool for technical support activities (Häll et al., 1983).

The computer hardware of the Measuring Computer System was renewed during the spring of 1987. The new system consists of a Norsk Data manufactured ND 110 which is a 16-bit general-purpose minicomputer, 1 MB of primary memory and a 28 MB Winchester type secondary memory. The system is integrated into the Local Area Network on site, which makes a very flexible use of the collected data possible. Graphic crt:s with matrix-printer hardcopy devices are utilised for users of the system.

During normal steady-state operation of the plant unit the measuring computer system is working in a disturbance recording mode of operation. This recording will be triggered by certain status changes of the protection system of the reactor or the turbine and also when a rapid change in electrical power output of the unit takes place. The history of the chosen signals are saved on a disk file covering the time before the triggering disturbance and after that, the same signals for a time interval after the disturbance. When the computer has finished the data collection, the data can immediately be presented and outputted together with appropriate scaling done. These plottings together with the output from the event recording function are one of the most important tools in analysing plant unit disturbances. This system has been operating from the time of the start-up of the plant units, and the collected data altogether gives a very good picture of the dynamic behaviour of the plant processes which can be fully utilised in analysing "new" disturbances.
When the disturbance recording function is active, other measurements can also take place. Process signal noise is measured and analysed as a routine 3-4 times a year. These analyses aim to give an early warning of beginning degradation of the analog controllers and actuating devices of the plant and also to give valuable information on the hydrodynamical state of the core. Noise spectra are compared to spectra collected as a 'fingerprint' of the plant processes.

4.3 Vibration monitoring

Another dedicated computer system is the turbine/generator vibration monitoring system for the surveillance of the mechanical state of the turbine and generator. The system can also be used for collecting data utilised in computing how to improve the balance of the axis by supplementary weights. The system collects vibration data from about 60 vibration monitoring transducers during each revolution of the axis and presents the data collected, fast and slow trend data, and gives alarms when certain preselected triggering levels have been exceeded. The collected data can be further analysed and outputted onto a matrix printer with graphical capabilities. The turbine vibration monitoring system has also been operating since the start-up of the plant units and the collected databank provides a very useful bulk of experiences.

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UTILITY EXPERIENCE IN REACTOR NOISE ANALYSIS IN GERMAN LWRS

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ABSTRACT

Reactor noise analysis has a long history in Fed. Rep. of Germany. Since the early 60ies it was directed to early fault, malfunction and anomaly detection in nuclear power plants. When successful diagnoses showed the high potential and benefits including economic and safety aspects, also the utilities began to have a vivid concern in application of the developed methods and systems. The know-how transfer from specialized teams to the onsite personnel as well as the development of more "intelligent" online systems are important present and future tasks for further improvement of their efficiency. The diagnostic group of GRS is active in both fields and has close cooperation with several utilities to achieve these objectives. Modern signal processing equipment and access to a comprehensive knowledge base stored in data banks are provided by the data analysis center of GRS. The available capabilities are described and examples of analysis results within the cooperation utilities/GRS are given.

KEYWORDS

Early failure detection, noise analysis, technical diagnostics, vibration monitoring, loose parts monitoring, operational experience.

1. INTRODUCTION

Taking into account the development of reactor noise analysis in many countries it has been decided by the international organizing committee for the 5th Symposium on Reactor Noise in Munich (SMORN V) to emphasize at this conference again, perhaps more than in former meetings, the operational experience gained by the practical application of these (relatively) new techniques in nuclear power plants. In special sessions representatives of utilities of the various countries should demonstrate their current status and their experience in order to enable the participants to obtain an overview of the worldwide application of reactor noise analysis and to be able to evaluate the contribution to the operational safety and reactor availability when using the new techniques.

For Germany (FRG), representatives of the nuclear power stations BIBLIS Unit A (4-loop, 1200 MW-PWR) and BIBLIS Unit B (4-loop, 1300 MW-PWR), OBRIGHEIM (2-loop, 350 MW-PWR) and NECKARWESTHEIM (3-loop, 900 MW-PWR) agreed to prepare this status report together with the GRS Garching, a research and development institute and their partner already for many years in the cooperation in the field of reactor noise analysis. Of course it is obvious, that primarily the situations in the own plants are described as an example. But differences in other plants might only exist as fas as the depth and detailism of the analysis are concerned.
2. HISTORY OF REACTOR NOISE ANALYSIS IN THE FED. REP. OF GERMANY

Early fault detection in the primary circuit of nuclear power plants and therefore reactor noise analysis have a long history in Germany (FRG). After the work in the 60ies concerning BWR stability and reactor frequency response (research institute LRA at the Technical University Munich, the predecessor institute of the GRS) and zero power reactor noise (Research Center Karlsruhe), it was started in 1969 in a cooperation of KWU and GRS (at that time Siemens and LRA), to investigate whether reactor noise can be used for malfunction and incipient or early failure detection in the non-accessible area of the primary system of a nuclear plant. Preliminary investigations at the PWR OBRIGHEIM (KWO) were successful and pointed to information upon reactor internal vibrations in the noise of ex-vessel neutron flux chambers. The subsequent development can be seen in fig. 1 ((a) cooperation with different plants and (b) activities and highlights). At the NPF STADE (KKS) the signals of the preoperational vibration test program for the reactor internals (cold and hot functional tests) were analyzed in detail, in order to get the necessary knowledge base for the reactor noise analysis. Using correlations between signals of vibration gauges inside and outside the reactor vessel (at the lid screw) and correlations between outside vibration gauges and excore neutron noise, for the first time the experimental proof for the core-barrel vibration as a source of excore neutron noise could be given /1/. In STADE, also for the first time, a loose parts monitoring system (LPMS) based on accelerometers has been installed and investigated (in a cooperative venture led by the Allianz Zentrum für Technik, AZT). GRS led a similar cooperative venture for the first LPMS in the BWR BRUNSBUETTEL. Briefly before at the BWR LIN-GEN, steam void velocities could be determined from incore neutron noise /2/ and signal sources in the neutron noise could be separated using the local-global concept /3/.

Fig.1a: History of utility/GRS cooperation in early fault detection

Fig. 1b: Activities and highlights within this cooperation

Further participations at the preoperational vibration tests in the prototypes BIBLIS (KWB-A) and NECKARWESTHEIM (GKN I) led to the knowledge base for the new designs of 1200 and 900 MW. Eventually at GKN (again together with KWU) a prototype of a reactor vibration monitoring system (VMS, in German SUS) has been installed. A theoretical structure model for a 3-loop primary circuit was developed and verified by experimental investigations /4/. Long-term investigations in GKN and later also in BIBLIS, Unit A and Unit B, gave important indications to operationally caused trending of particular features in the noise (vibration) patterns. For the newest 4-loop 1300 MW-PWR design a structure model was developed together with KWU and verified by preoperational hot functional tests and operational measurements at PHILIPPSBURG (KFP II) /5/. Just recently, a remotely controlled fast data transmission line for VMS-signals has been established between GKN and GRS and tested in order to investigate how a central data bank and knowledge base can be used in the best way for the particular plants.
All these programs and projects have been sponsored for the most part by federal government ministries. The Ministry of Research and Technology (BMFT) especially sponsored the projects for model development and computer-based pattern recognition. The Ministry for Environmental, Nature and Reactor Safety (BMU) sponsored the method developments and measures for improving noise analysis techniques for applications in licencing procedures. Without their far-seeing projects certainly the present status of application in FR Germany would never be possible.

3. STATE OF APPLICATION IN GERMAN NUCLEAR POWER PLANTS

3.1 Systems and their Objectives

The basic work described above and the successful tests of the prototype systems led soon to the general use of LPMS in all LWRs and to the general use of VMS in almost all PWRs. For loose parts monitoring, the systems working in the audible frequency range and with accelerometers as sensors were successful and are used now in all plants. In the most cases the sensors are mounted by magnets at the surface of the pressure-retaining boundary. In some cases, however, the sensors are mounted with stretch-tapes (e.g. at nozzles). Nowadays also bolted sensors are used. The number and positions of the sensors at the reactor vessel and (for PWRs) at the steam generators were reported often in publications such as the SMORN V-contribution of Olma, Schütz /6/. In this paper, the state of application is described and examples for practical diagnosis results with impact analyses in NPPs are given. The same measurement technique as used for loose parts detection was also applied, with removable sensors, for early failure detection on other components like valves, check valves, pipe supports, steam generator tubes and even on fuel assemblies during the refueling (see below).

The VMSs are genuine multi-sensor-systems consisting of a combination of vibration sensors (absolute and relative), excore neutron detectors (noise) and dynamic pressure gauges in the coolant. The sensor positions and the their numbers (together more than 30) have been reported very often in publications such as in the SMORN V contribution of v. Niekerk, Sunder /7/. The knowledge on the meaning of the various frequency peaks is very high meanwhile, so that many components can be monitored. Using sensitivity studies with the theoretical models being developed, predictions of feature variations (deviations from normal) can be calculated for assumed failure mechanisms (see SMORN V contribution Bauernfeind /8/). In some plants, in addition to the so-called VMS (SÜS) signals dynamic signals like incore neutron noise, lowpass-filtered LPMS-signals or, nowadays in all plants, shaft vibration signals of the main coolant pumps, are integrated in the vibration monitoring programs.

Other investigations which have applied noise analysis techniques and have been performed in FRG are: leakage detection particularly on acoustic basis, crack detection (stress wave emission) and sensor failure detection (signal validation). For the first two objectives, KWU has developed advanced systems as part of an overall system including LPMS and VMS (see SMORN V contribution Jax, Ruthrof /9/). For sensor monitoring, an universally applicable µP-based monitoring systems can be applied as described in the SMORN V contribution of Saedtler /10/.

3.2 Guides and Standards

The early realization of the developed measuring methods for early fault detection to operationally usable systems has been influenced certainly by the fact, that relatively soon (in 1974) in the RSK Guidelines (RSK = Reactor Safety Commission of the BMU) recommendations were formulated,

- to detect loose and detached parts in the reactor vessel
- to locate the loose parts as good as possible
- to measure the vibration behaviour of the reactor internals during the commissioning phase and
- to take care that these measurements can be repeated.

In the year 1984 the standard KTA 3204 of the Nuclear Safety Standard Commission was published. The standard deals with the reactor internals and asks for LPMS and VMS as systems which have to be available in the nuclear plants. Requirements are included for repetitive measurements and analyses (three times per fuel cycle).
For the design and the monitoring programs with LPMS, a DIN standard (25475 part I, DIN = German Industry Standard) exists since 1983. An analogous standard is under work for vibration monitoring. It should be mentioned that the LPMS-DIN-standard was used as a basis for the international IEC standard on LPMS, which is circulated now as a Central Office paper (IEC SC 45A CO (119)), that means it is asked for acceptance and, if so, will be published in the near future.

3.3 Utility Engagement in Early Fault Detection Systems

After initial skepticism, the early failure detection systems have been accepted by the utilities when failures which occurred in some plants were successfully diagnosed with these systems. In all nuclear power plants, a specific engineer is nominated who is responsible for LPMS and VMS. He has to decide, whether an alarm is "real" or is due to failed instrumentation or due to a known reason. The repetitive measurements and analyses as required in KTA 3204 are performed in some cases by own personnel of the utilities (spectral analyzers are available in the VMSS); most utilities, however, cooperate with specialized teams or contract for the required measurements. It is obvious that sometimes noise analysis is seen from plant operators as an unusual and relatively time-consuming procedure and therefore more automation, up to real online-diagnosis systems, is desired to monitor all the components automatically and to present the diagnosis results as a clear text at a display to the operator. Nevertheless, in some plants there are already interested people with high personal engagement which are able to make their own analysis and diagnosis.

4. KNOW-HOW TRANSFER TO AND SUPPORT OF THE UTILITIES BY GRS

After the development of the basic methods, it was the particular concern of GRS from the beginning to get the methods to a wide acceptance and to a broad application practice. Therefore, within the research foundation of the VGB, Essen (Vereinigung der Großkraftwerksbetreiber), the head organization of the NPP utilities, R&D projects in several plants were started with the aim to carry on the development practice-oriented, to solve new problems in the plants with comparable techniques, to assist the utility in fulfilling the KTA standard and to transfer the know-how collected at GRS as good as possible to onsite (since there, it might be needed for fast response and extensive measurement campaigns should be reduced to special cases). In other plants with real failures in the past and involvement of GRS at that time on behalf of the authorities or the TUVs (technical consultants) GRS is continuing to perform the analyses.

4.1 Data Bank for Noise Analysis Signatures at GRS

From all the plants shown in fig. 1 over the years GRS has collected an enormous number of noise signatures of plants with different size and power, of normal and anormal conditions, of operationally influenced or failure caused deviations, of impacts due to loose parts, of trends etc. For vibration monitoring and loose parts monitoring, separate data banks have been established enabling a fast access for comparison purposes. As a consequence to the ongoing analyses, these data banks in the GRS analysis center are continuously supplemented. The gathered knowledge base can be used to consult the utilities (or authorities) immediately in arising problems as well as for the current research activities directed to the development of automated knowledge-based diagnosis systems.

4.2 Data Link to GRS

Presently there are 3 modes to transfer data to the analysis center of GRS in Garching/Munich, i.e. 3 ways to use the knowledge base in the data bank and to involve the specialized team of GRS (fig. 2).

- In path 1, the so-called standard analysis of GRS, the unreduced data are synchronously stored with multi-channel tape recorders. For vibration signals, PCM-storage (pulse code modulation) at one magnetic tape, presently up to 56 channels, is used (for acoustic signals, FM-storage (frequency modulation) with 14 channels). The great advantage of a synchronous storage of all signals of a multi-sensor-system is, that all signals represent the same process condition and no uncertainty due to sequential records does exist (fig. 3). In the laboratory of GRS the signals of the PCM-tapes are prefiltered, combined to sets of 32 channels and transferred to IBM standard label tape. At the main institute computer (Amdahl 5870), the program packages IRAX are available.
which allow the calculation of all auto- and cross functions with just one computer run. The results are stored at a matrix tape in the database. The analyst can display all the necessary functions at a CRT, superpose the functions and interpret any deviations. The desired functions can be plotted.

Fig.2: Three ways of data transfer to the GRS analysis center

Path 2 was realized just recently. Between GKN and GRS, a test- and demo-system for remotely controlled fast data transmission has been established using the public telephone network (fig. 4). The Datex-L line of the telephone company (Deutsche Bundespost) allows a transfer-rate of 9600 baud. 2 to 8 channels can be transferred synchronously with sufficient frequency range. The utility partner at the plant has to close connectors at a switching board and the transfer of the desired channel combination can be started from Garching. The data arrive via necessary format interfaces the host computer in GRS (HP 1000), where analysis program packages are available. The quality of the spectra has found to be excellent and in high degree comparable with the IRAX-results.

Fig.4: Fast data transmission system for vibration signals
Path 3 is also a new possibility and means installation of "intelligent" systems onsite with monitoring and diagnosis capabilities and only transfer of reduced data to the GRS analysis center (with tape cassettes, telefax, etc.). For this purpose, the condition monitoring system COMOS has been developed by GRS (see /7/ and fig. 5) and firstly tested at GKN. When in Grafenrheinfeld (KKG) the shaft of a main coolant pump (MCP) was ruptured and the need for a monitoring system with sufficient early prewarning capability came up, i.e., the necessity of frequency-selective analysis and monitoring, the COMOS solution found a high interest, since the need became evident for trending and monitoring in different frequency bands, for suited display and plot formats allowing visual correlations and for an alarm module being programmable to follow known rules. Two further plants, KKG and ISAR II, gave an order for a COMOS system.

4.3 Cooperation NPP Obriegheim / GRS

Survey. Obriegheim (KWO, built 1967) is the oldest German NPP and therefore highly interesting for investigations in the field of early failure detection and online-assessment of components. In 1982 the old cooperation, as started in 1969, has been renewed. At this time only the accelerometers of the LPMS at the reactor vessel, two excore neutron chambers and about 35 incore neutron detectors were available. Absolute displacement gauges at the reactor cover were installed meanwhile. GRS performs two or three standard analyses per year from the available signals. When the MCP shaft problem came up also relative shaft vibration signals were investigated.

The most important tasks up to now have been: location of metallic impacts within the reactor vessel, variation of coolant flow or coolant velocity respectively in the core due to changed fuel assembly geometry, identification of the thermal shield shell mode, behaviour of the core barrel (CB) beam mode after repairs at the hold-down springs (see below).

Example of diagnosis results at KWO. In the beginning the correlation analyses at KWO were very limited due to the lack of absolute displacement sensors at the RPV-cover and the non-accessibility of the most excore neutron detectors. Nevertheless, by analysis of lowpass-filtered LPMS-signals supplemented by neutron noise information, we were successful to identify the CB beam mode showing a slight trend to lower frequencies (see fig. 6) over the years from the 13th to the 17th fuel cycle. In two measurements during the 17th cycle still a small frequency shift of the peak with increase in amplitude could be observed (fig.7).

![Fig.6: RPV-CB frequencies behaviour during the last six fuel cycles of KWO](image1.png)

![Fig.7: RPV-vibration spectra using a LPMS-sensor (first and second measurement in the 17th fuel cycle)](image2.png)
When the internals were inspected, the points of support of the hold-down springs at the upper core structure (fig. 8 and fig. 9) showed material impressions. In dependence on the depth of the surface deviations, shim plates were underlaid below the spring blades (see fig. 10) in order to get the original spring force. After the restart in the 18th cycle, the CB beam mode as well as the CB-RPV pendular vibration showed frequency shifts in the expected directions. This can be seen for instance from fig. 11 showing the comparison between the 17th to the 18th cycle. Two measurements in a stationary status of the 18th cycle confirmed that the frequencies are stable now, i.e. the repair measures were successful.

![Fig.8: Upper core structure of KWO](image1)

![Fig.9: Hold-down spring at the upper core structure of KWO](image2)

![Fig.10: Position of hold-down springs and LFMS-sensor and thickness of the shim plates](image3)

![Fig.11: RPV-vibration spectra before and after the hold-down spring repair](image4)

4.4 Cooperation NPP Biblis / GRS

Survey. Already 1974 in BIBLIS A a detailed knowledge on the internal vibration behaviour could be collected by the participation in the preoperational vibration functional tests. Long-term investigations and comparison with results in other plants of different size were performed in the following time. When 1978 cracks in some CB-screws led to a gap in the core liner, GRS has been involved by the authorities to control any unallowed resonance shifts within the repetitive measurements following the KTA 3204 standard. Some time later also BIBLIS B was included in these measures. To be able to cover the tasks, absolute displacement gauges were supplemented in both units.
The repetitive analyses comprise ex-core neutron noise and RPV absolute displacement signals. In addition all the available in-core neutron noise signals and LPSM-signals were included. Special programs were performed for the location of intensive single sound events during the repetitive tests of the emergency and heat removal pumps (caused as we found by impacts of the pipes in their supports due to collapsing steam), for sound monitoring on steam generators and for analyses of direct vibration measurements of SG internals. Nowadays the analyses of MCP shaft vibrations were added.

Examples of diagnosis results at BIBLIS. Very impressive have been the diagnoses from vibration and neutron noise signals at Unit A, where, several months before the next refueling, failures could be predicted for the hold-down springs at the RPV flange /11/. The same holds for loosening of some screws of the secondary core support in BIBLIS B. Both predictions were confirmed by the inspections.

Two other interesting results shall be described here. The systematic trend analysis of various peaks in the spectra of BIBLIS A showed a remarkable behaviour of the CB beam mode over the core cycle. The peaks at ca. 7-8 Hz are very clear in the beginning of the cycle (BOC), are distinctly reduced in the most cases in the middle of the cycle and disappear more or less in the slope of the RPV-CB pendular mode at the end of the cycle (EOC), see fig. 12. A corresponding behaviour can be observed in the coherences. The behaviour was comparable from fuel cycle to fuel cycle, the inspections were without any findings. Reasons for this behaviour may be the elongation of the fuel pins during the burn-up leading to a distortion of free CB vibrations due to contacts with the upper core plate. The example shows that a typical resonance trend behaviour might be observed, however, it should be not misinterpreted as a fault indicator.

![Fig.12: Variations of the CB beam mode during the fuel cycles in BIBLIS A](image)

![Fig.13: Trend analysis of CB-shell mode vibration using coherence of "adjacent" ex-core neutron noise](image)

The second example (see fig. 13) shows the trend analysis of the shell mode of the CB over the last seven years (26 measurements). One of the main tasks was the detection of a greater failure of CB screws by means of CB-shell mode monitoring. The relevant peak in the coherence functions of adjacent ex-core neutron detectors (good coherence, opposite phase) has been monitored with a statistical discriminant using a COMOS work station. Remarkable is a slow sinking of the shell mode frequency at ca. 18 Hz, which is reasonable and can be generally observed in technical systems subjected to mechanical and thermal loadings during stationary and transient operation. Since the speed of the feature deviation is small it can be regarded as to be non-critical. In addition, due to a superimposed trend in the reactor power (right handside), a part of the frequency shift can be related to this influence (the measurement 7b/2 should not be considered, because the fuel management was very different at that time). The example shows that the not-easy determination of the CB shell mode can be trended very well by the discriminant method.
4.5 Cooperation NPP Neckarwestheim / GRS

Survey. The most contacts between a plant and GRS are with GKN. After the preop tests, 3-loop model development and verification by operational measurements, the prototype VMS has been installed (together with KWU). Long-term trends delivered important and detailed experience. Already very soon, the MCP shaft vibrations and the incore neutron noise were integrated. Removable sound instrumentation was used successfully to identify impacts in valves, check valves and even during the refueling (damage at the fuel assemblies, see below). For the first time in FRG a fast data transmission line for vibration signals has been installed which can be activated very quickly, if there is an actual demand. COMOS has been developed for continuous monitoring and trending of frequency-selective features. The utility made best experiences with this system. No special computer experience is needed, softkeys for all necessary functions and auto-start routines are available. There is an intensive contact of the utility with GRS using telefax for plots and tape cassettes for long-term trend exchange.

Examples of Diagnoses Results at GKN. As a consequence to the many activities in the cooperation between GKN and GRS, a great number of interesting insights in the reactor noise behaviour has been possible and will be possible in the future (e.g. the operationally caused signature deviations which are stored during the COMOS monitoring, interpreted by the operator and used for the knowledge base). Here just a selection of interesting results shall be described:

- Acoustic monitoring during the refueling
  A system (BUS) was developed /12/ which enables secured information on damages at fuel assemblies due to contacts with neighbour assemblies during refueling (unloading and loading). A combination of acoustic signals, underwater signals and the force at the lifting lug of the refueling machine is used. By correlations of events, damaged assemblies can be clearly identified. In fig. 14 examples for damages at fuel assembly spacer grids are shown which were identified by the method.

Fig.14: Defects at spacer grids of fuel assemblies generated during refueling and detected by acoustic monitoring in GKN

Fig.15: Deviations of RPV vibration spectra one month after a two-loop-operation at GKN

- Feature deviations after a two-loop-operation
  The vibration spectra at the RPV cover showed deviations of the CB beam mode and the RPV-CB pendular mode after a two-loop-operation caused by a failed MCP motor (see fig. 15). The relevant power diagram and the dates of the measurements can be seen from fig. 16. As shown in fig. 17 in a two-loop-operation an additional static load of ca. 120 kN results for the CB. As a consequence the clamping conditions of the system RPV/CL/upper core structure was changed. After four months, the most frequency deviations were reduced (fig. 18) and only the CB beam mode showed a small frequency shift. After seven months, i.e. also after the refueling, again an identical vibration behaviour can be realized (fig. 19).
Fig. 16: Reactor power diagram and dates of the measurements

Fig. 17: Additional static load to the core barrel during the two-loop-operation at GKN

Fig. 18: Deviations of RPV-vibration spectra four months after the two-loop-operation

Fig. 19: Spectra comparison after refueling seven months after the two-loop-operation

- Detection of failed hold-down springs
  In the last (11th) fuel cycle from the first to the second measurement (time interval 5 month), a distinctive frequency shift (0.3 Hz) of the CB beam mode was detected, but now for the normal 3-loop-operation. The prediction of reduced forces (about 10%) of the hold-down springs at the reactor vessel flange was exactly confirmed during the last refueling in Aug. 1987. About 10% of the 100 springs were defect.

- Signature deviations after bearing exchange in a main coolant pump
  After a refueling, distinct deviations of peaks in the spectra of the relative vibration signals in one loop were observed (fig. 20), especially for 10.9 Hz. Since also in the MCP shaft vibration spectra clear differences were found (fig. 21), the utility was asked for recent major maintenance actions. By doing this the reason for the feature changes could be clarified immediately. During the refueling the bearings of the particular MCP has been exchanged. 10.9 Hz is the so-called "oil-whip" which is distinctly available for vertically supported shafts with bearing clearances, but is, in contrast to horizontally supported shafts, uncritical. It also can be seen, that the "oil-whip" frequency creates side band modulations at 13.9 Hz and 35.7 Hz around the nominal speed rate of the shaft. The example shows clearly, that the
Fig. 20: Main coolant pump vibrations relative to the building before and after an exchange of MCP-bearings.

Simultaneous observation of the R-signals of the VMS (loop/pump housing against concrete) and MCP shaft vibrations (shaft against pump housing) allows a much better interpretation and signal validation of a new MCP vibration behaviour. Two and seven months after the bearing exchange, both the R-signals and the shaft vibrations showed a stable signature (fig. 22 and 23).

Fig. 22: Spectra comparison of MCP-relative motions two and seven months after bearing exchange.

Fig. 23: Spectra comparison of shaft vibrations two and seven months after bearing exchange.

- Cause identification for amplitude variations by spectral analysis (MCP shaft vibrations)

After the ruptures of MCP shafts in Gösgen and KKG, all German PWR-MCPs were equipped with eddy current sensors to measure the relative shaft vibrations. The amplitudes are displayed at a strip chart recorder and monitored with respect to relative amplitude variations. Since for vertically supported shafts of this size no qualified shaft rupture monitoring systems are available, since operational influences sometimes have a strong influence to these amplitudes and since the amplitude increase due to a shaft crack propagation may happen within very limited time, the knowledge of causes for amplitude variations and the availability of early warning indicators for a severe crack propagation play an important role. Fig. 24 shows clearly that for an increase in the record of a shaft vibration amplitude immediately the cause can be identified by spectral analysis (here shown in a waterfall representation).
The amplitude increase is practically completely caused by an increase in a frequency band which is related to the influences of the bearings. A timely correlation of these with COMOS stored and afterwards plotted spectra was easy: a secondary oil pump was switched on due to reactor protection tests.

Fig. 24: Advantage of spectra analysis for the interpretation of amplitude changes in the strip chart records of MCP shaft vibrations

5. SUMMARY AND OUTLOOK

The given examples show impressively that the dynamic measures (noise signals) as gained by modern early fault detection systems in the German nuclear power plants make available a plenty of useful information needed operationally (e.g. maintenance management) or for safety aspects (licensing requirements). An important task for specialized teams is to enable the utilities by an intensive knowledge transfer to the onsite personnel, to perform their own assessment and evaluation. GRS is active in this sense and is able to assist the utilities with its facilities and knowledge base. With new systems like the user-friendly COMOS, a lot of analysis work can be automated onsite and also (first) classification can be performed. Further developments of systems like COMOS /7/ or SPIRIT /9/ to real diagnosis systems, i.e. the development to artificial-intelligence-systems relating actual signatures to the collected and still growing knowledge base, will bring the necessary release of the plant operators.

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LOOSE PARTS

Session chairman: F. Akerhielm (Sweden)
In this session there were four papers. Three of them dealt with loose part monitoring and one paper dealt with leak detection.

The first paper by Mayo and Shugars, presented by Mayo, shows results from a research project to use metal impact theory, plate wave transmission theory, and experimental data to develop an analytical basis for loose part monitoring system performance. The results can be used to design or evaluate loose part monitoring system detection and diagnostic capability. They also provide a basis for using either known energy or measured force impact testing to calibrate system response over the range of potential impact signals. The findings of this work have been translated into guidelines for American utilities to use in defining, evaluating and improving loose part monitoring system performance.

The second paper by Castanie et al., presented by Lehmann, reported on recent experiences in loose parts monitoring of light water reactors. The examples given demonstrate a whole spectrum of applications and advantages of the loose part monitoring system installed in all PWRs and BWRs in the FRG. A new burst analysis method of sound mode separation permits the localization of the source anomaly with few, if necessary only one, accelerometer signals.

The third paper by Olma and Schutz, presented by Olma, described advanced burst processing methods in loose part monitoring. They showed in examples that, from the view of early failure detection, the methodology of acoustic monitoring covers important kinds of mechanical contacts at primary systems both in PWR and BWR. Increasing plant age and enhanced use of digital analysis systems will intensify the trend to condition monitoring of actual component status. Access to a specialized knowledge base is expected to improve the interpretation of the bursts.

The fourth paper by Brunet et al., presented by Kong, described an acoustic method for leak detection in a steam generator of SUPER PHENIX. The role of the acoustic system is to enable fast action in the event of a leak growing rapidly which could rupture neighbouring tubes. The simulation of water leaks in the steam generator by argon injections performed to date at 50% of the rated power has shown promising results. The observed decreasing signal-to-noise ratio with increasing source-sensor distance is expected to give information on the axial localization of leaks.

The papers in this session illustrate the use of acoustic signals both for loose part monitoring and leak detection. The US paper on LPM is an approach from the theoretical side with comparisons with experimental data, while the FRG paper concentrates on experience from the use of LPM. The papers show that LPM has now reached a mature state both from the theoretical and the application point of view. It is clear that the equipment installed on most plants now can be used efficiently. The LPM systems now evidently contributes substantially to the availability of nuclear power plants. Also the cost/benefit equations have proven to be favourable. The French paper on leak detection in SUPER PHENIX also shows how theory and experience can be successfully used for applications.
LOOSE PART MONITORING SYSTEM IMPROVEMENTS

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Abstract - Metal impact theory, plate wave transmission theory, and experimental data were used to develop an analytical basis for loose part monitoring system performance. Valid signal properties were defined and guidelines were developed for the number and location of sensors, sensor mounting, calibration methods, signal filtering, and interpretation.

1.0 INTRODUCTION

A research project, sponsored by the Electric Power Research Institute (EPRI), was performed to develop a general technical basis and guidelines for utilities to use in specifying, evaluating, and improving loose part monitoring system performance. This work included the development of methods and guidelines for specifying the number, location, and mounting of sensors, selecting and applying calibration methods, defining alarm set-points, and diagnosing the size and location of a loose part. This project was performed in cooperation with Duke Power Company and the Tennessee Valley Authority. Additional technical data and review were provided by Northeast Utilities and the Gesellschaft Fuer Reaktorsicherheit.

Laboratory and in-plant testing was performed to investigate the validity of Hertz impact theory over a range of loose part mass and energy. Metal sphere and force instrumented hammer impact data were acquired to demonstrate and validate models for plate wave magnitude and frequency content, distance attenuation, and dispersion. Detection sensitivity was defined as a function of impact mass, energy, and signal filtering for representative background noise. Sensor location, sensor mounting, and impact calibration methods were evaluated with respect to the ability to estimate impacting object mass, location, and energy.

The results of this work demonstrate the validity of Hertz impact theory over a substantial range of loose part mass and energy. Plate bending wave theory was shown to predict observed signal characteristics in good detail. The associated signal models and data provide an analytical basis for defining, evaluating, and improving loose part monitoring system performance, calibration, and signal interpretation.

This paper discusses technical findings regarding metal impact properties, initial plate wave frequency and amplitude, wave transmission, and signal filtering. Associated loose part monitoring system guidelines are summarized. These findings have been coordinated with the American Society of Mechanical Engineers subgroup that is preparing a standard on LWR Loose Parts Monitoring. The results of this project are expected to lead to improvements in loose part monitoring system capability and use.

2.0 IMPACT PROPERTIES

Hertz impact theory has been used to investigate metal impact properties for loose part monitoring applications for a number of years (Izumih, et al., 1985; Oima, 1985; Mayo 1985; Fujita and Tanaka, 1982). This theory defines the motion of the initial point of contact between a solid metal object and an infinite plate as a half sine function with half period and amplitude corresponding to the impact contact time and the peak displacement during the time of contact. These parameters are governed by the impacting object mass, velocity at the time of initial contact, and radius of curvature at the point of contact.

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The time history of impact motion defined by this model can be combined with Newton's second law of motion to predict the peak contact force and acceleration of the impacting object as a function of impact parameters. The resulting equations for peak displacement, velocity, acceleration, and force for the impacting object are:

\[ D_{\text{max}} = k_h \left(\frac{m v_o^2}{2}\right)^{1.2} R^{-2} \]  
\[ V_{\text{max}} = V_o = \left(\frac{v_1 R}{k_h t_h}\right) \left(\frac{m v_o^2}{2}\right)^{1.2} R^{-2} \]  
\[ A_{\text{max}} = k_h^{-1} m^{1.2} v_1 R^{1.2} \]  
\[ F_{\text{max}} = k_h^{-1} (m v_o^2)^{0.6} R^{-2} \]  
\[ k_h = \left(\frac{15}{16}\right) \left(\frac{(1-v_1^2)/(1-v_2^2)/E_1}{(1-v_2^2)/E_2}\right)^{1.4} \]

where

\[ V_o \] is the initial velocity of the object

and

\[ v_1 \] and \( E_1 \) are Poisson's ratio and Young's modulus for the plate

\[ v_2 \] and \( E_2 \) are Poisson's ratio and Young's modulus for the object

\[ m \] is the mass of the object

\[ R \] is the radius of curvature at the point of contact

The impact contact time can be similarly expressed as

\[ t_h = \frac{v_1}{k_h m^{1.2} v_o^{1.2} R^{1.2}} \]

These functions show that there is no linear relationship between the primary impact parameters of impacting object mass, velocity and contact radius, or momentum or energy, to the impacting motion displacement, velocity, or acceleration. In general, it is not possible to uniquely estimate the three impact parameters from the observed parameters of signal amplitude and frequency. For some shapes, such as spheres, the dominance of radius in the mass term permits the estimation of mass and velocity from observed parameters with reduced uncertainty. The Hertz impact equations provide a basis for investigating signal properties as a function of loose part shape, material, and velocity.

The derivation of Hertz theory involves several assumptions including requirements that the impacting object respond as a solid body, that there is no inelastic deformation, and that contact time is such that the impact primarily excites bending wave motion in an essentially infinite plate. Experimental work by others (Olma, 1985; Fujita and Tanaka, 1982) and measurements performed during this work show that these requirements are satisfied for a substantial range of loose part impacts.

Impact contact times predicted by Hertz theory were tested with a series of impacts using metal spheres weighing between approximately 2 ounces and 32 pounds (1" to 6" diameter). These impacts were performed against the side of a steam generator in a preoperational plant. The contact times were measured by an electrical circuit connected between the steam generator and the impacting spheres. The results of these measurements, presented as a function of sphere diameter and energy in Figure 1, show remarkable agreement between predicted and observed contact times over a wide
range of mass and energy. Subsequent laboratory testing demonstrated that most of the variance in the measured data was due to variations in contact time resulting from surface roughness. Laboratory results also demonstrated that Hertz theory is valid for impacts against a plate type structure as long as the plate boundaries or supports are far enough away from the impact point that the plate wave is not reflected back to the origin before the impact is completed. This distance is typically on the order of one foot and can be calculated from the bending wave velocity associated with the primary impact wave frequency.

Hertz theory is not limited by impacting object shape as long as the object responds as a rigid body during the impact and does not translate any energy into rotational motion. These requirements were satisfied by metal spheres and force instrumented hammers with hammer heads less than three inches long used during this work. Hammer head elastic wave participation in the impact response has been demonstrated for a hammer head that was approximately sixteen inches long (Fujita and Tanaka, 1982). Objects used in impact testing must have some radius of curvature to avoid errors that can occur due to initial contact on the edge of a flat surface.

3.0 PLATE WAVE FREQUENCY

Initial plate motion at the point of impact is translated into a cylindrical wave that expands away from the impact location. This wave has a different shape and frequency content than the initial half-sine motion at the point of contact. These differences were noted in early work on impact theory (Raman, 1920) that combined Hertz theory with transverse wave properties (Lamb, 1902). One difference between the plate wave and motion at the impact point is that the initial half period of the propagating wave is a factor of about 1.6 shorter than the impact contact time. The plate wave acceleration is not exactly sinusoidal and had significant amplitude for only a few cycles. As a result, the wave frequency spectrum has a peak near the frequency associated with a sinusoid that has a half period about 60% shorter than the contact time, and is spread over a band that is about +/- 50% of this frequency.

The shape of a typical initial plate wave acceleration signal, measured on a steam generator shell, is shown in Figure 2 along with the generating force signal. A typical frequency spectrum, obtained from the impact of a mass of 10 pounds, is shown in Figure 3. This spectrum shows the primary impact wave spectrum in the range from about 1 to 5 kHz, sensor mounted resonance response at about 15 kHz and sensor crystal resonance response at about 22 kHz. The impact wave frequency and the relative magnitude of sensor resonance response increases as mass decreases. Other examples of impact wave time and frequency domain signals are included in later figures.

![Fig. 2. Initial Impact Wave and Generating Force](image1)

![Fig. 3. Initial Impact Signal Spectrum - Ten Pound Mass](image2)

The combination of Hertz impact theory and plate wave theory can be used to predict primary frequency content of loose part impact signals as a function of mass and energy. The central frequency of loose part impact waves, as predicted from Hertz theory and Raman's work, is shown in Figure 4 for a range of metal spheres diameters and velocities representing a range of potential loose part mass and energy. This figure shows that most of the information from the impact of loose parts weighing between about 0.15 and 32 pounds and impacting with velocities between about 1 and 10 feet per second will be in the frequency range between about 1 and 10 kHz.

The ability to estimate the primary impact wave frequency was investigated by counting zero crossings in the leading edge of experimental impact waveforms. Good agreement with theoretical predictions was obtained for signals where there was not a mechanical resonance that dominated the observed frequency.
The frequency content of metal impact signals was also investigated by calculating the coherence function between impact force and detected acceleration. Good coherence was found over the band where primary wave frequency content was expected. An example coherence function for the impact of a one pound steel hammer is shown in Figure 5. The results for a 3 pound force instrumented hammer were similar except that coherence did not extend to as high a frequency. As part of this investigation, it was noted that substantial detected signal amplitude was also present above 10 kHz, although it was not coherent with the impact force. This higher frequency content was associated with mechanical amplification of higher frequency terms in the impact wave by the sensor mounted and crystal resonance frequencies. As demonstrated by Figure 5, the magnitude of this higher frequency signal content is not correlated with impact force due to irregular interaction between the shape of the impact wave spectrum and singular resonance frequencies. Impact signal frequency content is considered further in Section 6.0 SIGNAL FILTERING.

Fig. 4. Hertz Impact Plate Acceleration Wave Frequency

Fig. 5. Coherence Between Impact Force and Detected Acceleration - One Pound Steel Hammer

4.0 INITIAL PLATE WAVE AMPLITUDE

Lamb's solution for transverse wave shape, normalized to the peak displacement amplitude calculated from Hertz theory, can be used in principle to calculate the plate surface acceleration wave shape and magnitude as a function of impacting object mass and velocity. This is not easily done due to the nature of the equation. An alternate method to relate impact properties and wave shape to the magnitude of the plate acceleration wave is to assume that the acceleration wave consists primarily of a few half cycles at a frequency that is 1.6 times the frequency that has a half period equal to the impact contact time. This assumption models the major properties of observed signals. The magnitude of this signal can be normalized to peak plate acceleration calculated as

$$A_{plate} = \frac{F_{max}}{M_{eff}}$$

where

- $F_{max}$ is the peak impact force calculated from equation (4)

and

- $M_{eff}$ is an effective mass of the plate volume that responds during the impact contact time expressed as

$$M_{eff} = \rho_{steel} (C_b t_h)^2 h$$

where

- $h$ is the plate thickness
- $C_b$ is the bending wave velocity
- $\rho$ is the density of steel

The range in initial plate wave acceleration magnitude predicted by this method is shown in Figure 6 for spheres with diameter between 1 and 6 inches (mass between 2 ounces and 32 pounds) and impact velocities between 1 and 10 ft/sec. This figure indicates that a substantial proportion of loose part impact signals will range from about 1 to 100 g peak at the impact location.

5.0 PLATE WAVE TRANSMISSION

The magnitude of impact wave motion is a function of parameters including the dominant wave mode, spreading of the wave as it expands over larger distances, and losses that occur due to the radiation of acoustic energy into the surrounding media. Wave transmission theory provides a basis for identifying and evaluating these effects.
Hertz theory and experimental data show that the dominant wave mode for loose part impact signals in light water reactor pressure boundary components is the so-called "bending" wave mode. This conclusion is based on experimental agreement between predicted and measured impact contact times and the observed property of bending waves to propagate different frequencies with different velocities, leading to changes in the wave shape as a function of distance from the impact location.

Since bending waves propagate different frequencies at different speeds, and impact signals are transient, the observed time required for an impact wave to travel over a given distance reflects a "group" velocity that is a function of the dominant frequency term in the detected signal. Bending wave velocity is also a function of plate thickness, so that different speeds are observed in components of different thicknesses. A theoretical curve for bending wave group velocity as a function of frequency is shown for several plate thicknesses in Figure 7. Considering Figure 4, observed loose part impact signal velocities will range from about 5,000 to 10,000 ft/sec. Also, impact signals from masses less than about one pound will not exhibit substantial dispersion in thick plate structures due to the primary frequency content being in the asymptotic region of group velocity.

Bending waves are attenuated as a function of distance due to the loss of energy to the surroundings on both sides of the plate and due to increasing wave area as it propagates away from the impact location. Wave losses from the surface of structures to air are very small and need not be considered. Losses to water are relatively small for reactor vessel plate thickness but can become significant over distances of tens of feet in components less than about 4 inches thick. Example calculations for bending wave loss to water at 5,000 Hz are shown for several plate thicknesses as a function of distance in Figure 8.
The effect of increasing wave area is a significant attenuation factor. This attenuation can be expressed as a difference in Hankel functions of the second kind (Cremer, et al., 1973). There is a "near field" where attenuation is extremely rapid, and a "far field" that can be approximated by exponential or low order polynomial functions. Example curves for distance attenuation at 5,000 Hz are shown in Figure 9. The near field is strong within about 3 feet of the impact location. Outside of the near field, the wave is attenuated between factors of about 10 and 20 over representative distances between loose part impact and sensor locations. Distance attenuation is a function of both frequency and plate thickness due to terms involving bending wave velocity.

Fig. 9. Distance Attenuation Function for 5000 Hz

Both the models for initial wave magnitude and distance attenuation were supported by in-plant test data acquired on a steam generator shell. Examples of this agreement are shown in Figures 10 and 11 for impacts generated by 1 and 4 pound metal spheres. The solid line is the predicted acceleration signal magnitude for a sensor about 30 inches from the impact location. Experimental data are shown as points. These figures also demonstrate how smaller masses generate higher magnitude signals for equal impact energy. Similar curves were obtained for other masses.

Fig. 10. Theoretical and Experimental Response to Impact of One Pound Ball

Fig. 11. Theoretical and Experimental Response to Impact of Four Pound Ball

The range of loose part impact signal acceleration, estimated as between 1 and 100 g at the impact location from Figure 7 is extended down to about 0.05 g over distances that can occur in major reactor coolant system components. If sensors are placed so that signals can always be analyzed from sensors that are more than about 3 feet from the impact location, the near field can be avoided and the maximum signal magnitude is about 20 g.

These analyses are based on sensors that are mounted on and normal to one side of the impacted surface. Additional attenuation is present if the impact signal must travel through structural connections to reach the monitored surface. This is a complex problem for impacts against reactor or steam generator internal structures. Amplitude interpretation of signals from impacts that are
not directly against part of the pressure boundary will involve a high degree of uncertainty unless impact calibration measurements have been performed at the same point as the loose part impact. This type of detailed internal impact calibration is almost always not available due to the difficulty of acquiring this type of data.

Similar signal attenuation effects occur for sensors that are mounted on small diameter piping that extends from the major monitored component. These locations commonly include incore detector guide tubes below the reactor vessel, and in some cases, control rod drive tubes and small diameter piping leading from steam generators. Calculations based on the attenuation of bending waves at structural branch points (Cremer, et al., 1973) and data acquired during this program indicate that additional attenuation factors on the order of a factor of ten are associated with these sensor locations.

6.0 SIGNAL FILTERING
Cumulative frequency spectra were used to define the distribution of impact signal amplitude as a function of frequency band. An example distribution for the impact of a ten pound sphere, Figure 12, shows that the detected signal amplitude is divided between the primary wave frequency content between 1 and 5 kHz, sensor mounted resonance response at about 15 kHz, and sensor crystal resonance response at about 22 kHz. The frequency of the primary impact wave contribution and the relative magnitude of sensor resonance response increases as impact mass decreases.

Loose part monitoring system channel background noise amplitude distributions were measured for a pressurized water reactor under full power operation. A typical cumulative background noise spectrum obtained from a sensor that was drill and tap mounted into the edge of a steam generator tube sheet is shown in Figure 13. Most of the background signal amplitude is distributed between flow and vibration signals below about 1 kHz, the sensor mounted resonance frequency, and the sensor crystal resonance frequency. A small background at 5 kHz is also indicated. The background noise distribution for a sensor that was clamp mounted against an incore detector guide tube, Figure 14, has similar terms, except that the contribution from low frequency vibration is higher.

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Fig. 12. Cumulative Frequency Spectrum For Impact of Ten Pound Metal Sphere

Fig. 13. Cumulative Frequency Spectrum For Steam Generator Background Noise

Fig. 14. Cumulative Frequency Spectrum for Lower Reactor Vessel Background Noise
In each case detection sensitivity can be greatly enhanced through the use of bandpass filters between about 1 and 10 kHz. The 1 kHz high pass filter will remove low frequency flow and vibration signals without significantly attenuating impact signal frequency content for masses up to about 30 pounds. The 10 kHz low pass filter will remove background noise amplified by the sensor mounted and crystal resonance frequencies while not significantly attenuating impact signal frequency content for masses greater than a few ounces.

Increasing the break frequency of the low pass filter will increase the detected signal magnitude and detectability of loose parts that weigh below a few ounces. However, the ability to interpret signal amplitude and frequency content is limited by interaction of the impact signal spectrum with the sensor resonance frequencies. Systems with higher sensor mounted resonance frequencies can use wider filter pass bands to improve the sensitivity to parts less than a few ounces without loss of amplitude and frequency information to sensor resonance response.

As another example of impact wave properties and filtering, signals generated by the impact of two different masses against the side of a steam generator with no background flow noise are shown in Figures 15 and 16. In each case the top trace is the unfiltered signal and the lower trace is low pass filtered at 10 kHz. The smaller mass is seen to excite significantly more ringing of the sensor resonance, increasing the detected signal magnitude and obscuring the true impact waveform. The low pass filtered signals show the impact waveforms clearly including multiple arrivals as the waves circle the steam generator in clockwise and counterclockwise directions. It can also be noted that the higher frequency signal exhibits significantly less dispersion as a function of distance as predicted by the bending wave group velocity curve.

![Figure 15. Impact Signal for 0.2 Pound Sphere Against Steam Generator](image1)

![Figure 16. Impact Signal for Ten Pound Sphere Against Steam Generator](image2)

7.0 SUMMARY
Theoretical and experimental results of this research demonstrate that:

- Hertz impact theory is valid over a substantial range of loose part mass and energy.
- The initial impact wave has a half period that is about 1.6 times shorter than the impact contact time.
- The dominant wave mode for loose part impact signals in light water reactor pressure boundary components is the "bending" wave mode.
- Wave area expansion is the dominant attenuation effect and has a near field radius of about 3 feet.
- Most of the information from the impact of loose parts, within stated ranges of mass and velocity, lies between 1 and 10 kHz.
- Wave amplitude and frequency content are important for mass and energy estimation.
- Filtering signals below 1 kHz and above 10 kHz removes substantial background noise without attenuating impact signal frequency content for masses greater than a few ounces.
Hertz impact theory, wave transmission theory, and sensor response models can be used to define loose part signal properties in good detail. These models can be used to define the expected frequency content, amplitude, and wave shape of detected metal impact signals as a function of mass, energy, distance from sensors, and sensor frequency response. These results can be used to design or evaluate loose part monitoring system detection and diagnostic capability. They also provide a basis for using either known energy or measured force impact testing to calibrate system response over the range of potential impact signals.

Other findings include the importance of sensor mounting to obtain broad band uniform frequency response to the extent possible, the value of high pass filters to improve detection sensitivity by removing low frequency flow and vibration signals, the value of low pass filters to remove sensor resonance effects from signal amplitude and frequency content interpretation, and the importance of performing metal impact calibration measurements that include mechanical transfer functions in the definition of channel frequency response.

The results of this work have been translated into guidelines for American utilities to use in defining, evaluating, and improving loose part monitoring system performance. Technical recommendations from these guidelines include:

- The use of six sensors on pressurized water reactor vessels, three around the top and three around the bottom, to permit impact location, amplitude correction for distance attenuation, and improved rejection of sound conducted into the vessel by external piping
- The use of twelve sensors on boiling water reactor vessels, arranged in rings of three at four different elevations for the reasons noted above
- The use of at least two sensors at steam generator inlet regions for redundancy and for contribution to separation of primary and secondary impacts in U-tube generators
- The use of high resonant frequency sensor mounting techniques to the extent possible to preserve uniform response for a range of loose part mass
- The use of bandpass filters to reduce background noise and select frequencies associated with primary impact waves over the range of loose part mass of interest
- The use of a range of impact test mass to excite the range of potential loose part signals if known impact energy calibration is used
- The use of a range of hammer mass or frequency dependent transfer function calculations to measure channel frequency response if instrumented hammers are used for impact response calibration
- Interpretation of setpoints as a combined function of mass and energy
- Recognition of the frequency dependent velocity of sound and wave dispersion in impact location analysis
- The use of signal amplitude and frequency content for mass and energy estimation

Signal theory derivations, additional experimental data, and more detailed discussions of recommendations and guidelines will be published in the final report for EPRI research project RP2642-1.

REFERENCES

RECENT EXPERIENCES IN LOOSE PARTS MONITORING OF LIGHT WATER REACTORS

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Abstract - After a short description of the KWU-Loose-Parts-Monitoring-System (KUS) four examples of analysis of anomalies are described, which demonstrate the advantages of this surveillance:

- improvement of operation of the plant
- improvement of availability of the plant
- avoidance of further damage
- reduction of radiation exposure to personnel.

A new analysis method for localization of the sound sources is outlined, demonstrated by test impacts and successfully applied for one actual anomaly.

1. INTRODUCTION

KWU with 20 years of experiences in loose parts monitoring (LPM) offers not only LPM-systems but also services for the maintenance of the systems and assistance in case of alarms or anomalies.

All PWR's and BWR's in the Federal Republic of Germany are continuously monitored by LPM-systems called KUS (KörperschallÜberwachungssystem).

Fig. 1 shows a typical arrangement of the KUS-sensors for a 4-loop PWR. Accelerometers are located at the natural collection regions for detached parts. Additional accelerometers are mounted at the upper part of the RPV, where the sound coupling between RPV internals and the pressure boundary is high, and near the feedwater lines at each SG. The signals of the accelerometers are continuously monitored in the audible frequency range using both fixed and sliding thresholds.

![Fig. 1. LPMS: Typical arrangement of accelerometers](image)

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Alarms are annunciated in the main control room and simultaneously the signals from all surveillance channels are recorded synchronously by a microprocessor controlled multichannel transient recording system which offers special LPMS evaluation software, as well as facilities for data storage and hardcopy. KWU performs upon request special services with respect to the LPMS, such as:

- system functional tests including test impacts
- assistance in case of alarms of recognized anomalies, e.g.
  * interpretation of signals and data
  * determination of possible source of the anomalies
  * additional special LPMS surveillance techniques
  * recommendation of further procedures.

In the past years KWU was called to several plants after alarms from the KUS. Some examples of structure born sound anomalies are discussed below.

In order to perform these services KWU also investigates basic properties of sound generation and propagation. One important aspect is the localization of the sources of sound anomalies. A new method has successfully been applied for localization, which uses the dispersion behaviour of structure born sound waves.

2. EXPERIENCE WITH LPMS

2.1 LPMS alarms of steam generator channel

KWU was informed by telephone that a high burst rate occurred at an accelerometer near the feedwater line of a steam generator (SG) and the strip chart recording shown in Fig. 2 was transmitted to KWU by telexcopy.

![Figure 2: Bursts in a signal from the steam generator](image)

A: Steam Generator 2 near feedwater line
B: Steam Generator 2 entrance chamber

Fig. 2. Bursts in a signal from the steam generator

It shows burst signals from metallic impacts with varying amplitudes and time intervals.

Since a localization from differences in delay of bursts was not possible because the anomaly exhibited itself only in the signal of one sensor, a rough estimate relied on the very steep rise times indicating that the source of the sound was rather near the sensor.

Since from the information available it was not clear, whether that anomaly was due to impacts at the SG itself including its internals or due to impacts on the secondary side KWU proposed the following actions and procedures:

- Change of operational conditions, especially operating auxiliary systems and subsystems with piping near the main feedwater line.

- Attachment of an additional accelerometer magnetically on the feedwater line in the accessible region in order to determine the origin of the sound and its propagation direction. This would have decided whether the anomaly occurred within the SG or not. Also the distance of the anomaly from the sensor could possibly be deduced.
The customer followed the KWU proposal and first changed the operating conditions. This already demonstrated the source of the bursts to be generated in a valve in an emergency feedwater line, which was opening and closing in an uncontrolled way, because of a lack of pressure difference.

After the pressure had been released at one side of the valve, the valve closed and the sound anomaly stopped. This is a nice example, where the KUS helped to avoid damage to a gasket on the secondary side.

2.2 Loosened bolt in a SG

In another plant similar alarms as described above occurred. Therefore also the same actions were proposed. In this case change of operating conditions did not lead to the desired result. With the additional sensor on the feedwater line correlated bursts could not be detected. Therefore the additional sensor was adapted also at the SG boundary opposite to the normal LPMS sensor. With this sensor configuration about 50 correlated bursts were observed within a day with special surveillance equipment. One example of the bursts acquired is shown in Fig. 3.

The analysis by KWU personnel revealed that the amplitudes and time differences of the signals were always equal within certain limits. It was deduced that a loosened part existed. An area of possible locations could be defined.

During refuelling the inspection of this sector revealed a loosened bolt in the support of the water separator.

2.3 Detached part in SG entrance chamber

KWU was called to a plant where during the start up operation (about 25% power) after refuelling extremely large burst signals at the entrance chamber of a SG occurred 3-4 times per second.

The signals of the accelerometer at the preheating chamber of the steam generator also showed bursts correlated to those at the entrance chamber.

In Fig. 4 some bursts are displayed which had maximum amplitudes of up to 30 g (g = earth acceleration).

The bursts in the entrance chamber signals were extremely steep, resulting from impacts very near to this accelerometer. As the time lags between the rising edges of the bursts in the SG channels were varying a detached part was indicated in the entrance chamber of the SG hammering against the walls at different locations.
From the high accelerations measured the mass of the loose part was estimated to be at least 100 grams.

![Graphs showing acceleration and time](image1.png)

*Fig. 4 Burst in signals from steam generator sensors above: sensor at preheating chamber below: sensor at entrance chamber*

Following the KWU proposal the plant was shut down immediately.

During the switch off of the last pump the bursts were monitored, tape recorded and analyzed. The time lag of the last burst between the two surveillance channels unambiguously showed that the detached part had remained in the SG entrance chamber and not fallen into the main coolant pipe. The detached part was secured where predicted and identified as a broken pin from the core barrel flange (Fig. 5) with a mass of 197 grams. This bolt serves as centre bolt for the first mounting of the upper core structure during commissioning and is not necessary for plant operation. An inspection of the SG entrance chamber revealed that severe damage had been avoided because of the early detection of this broken bolt.

![Diagram showing detached part](image2.png)

*Fig. 5 Detached part found in the SG: bolt from the core barrel flange*

3. IMPROVEMENT OF LOCALIZATION METHOD

The localization of the source of a sound anomaly using accelerometer signals can be very difficult in practice. As shown in our first example the anomaly may show up in the signal of a single accelerometer only.

In other cases where bursts in several monitoring channels appear, the localization methods using time lags between bursts in different signals have sometimes been rather inaccurate and generally can only deduce the point where the sound
reaches the pressure boundary but not the sound source itself. Recently a new method has been developed by KWU, which enables the deduction of the distance to the sound source by taking advantage of the dispersion behaviour of structure born sound modes /1/.

Fig. 6 shows the low frequency part of the wellknown lamb diagram of plate waves in steel.

Fig. 6 Part of the lamb diagram of plate waves in steel

As bursts are wave trains consisting of a frequency mixture there is the possibility of separating the a₉- and s₈-mode components which have very different velocities in the low frequency region, and to try to use this for localization as has been shown in /2/.

Our method uses narrow-band filtering of the burst signals. There the time lags between the rising edges of the different burst-components as a function of the different filter frequency bands are deduced. If D+L is the distance from the sound source (in the general case D and L are distances in structures with different wall thicknesses), the time lags Δtᵢ between the burst components at the frequency bands fᵢ and fᵢ can be written as

\[ Δtᵢ = tᵢ - tᵢ = (D+L) \left( \frac{vᵢ - vᵢ}{vᵢ vᵢ} \right) \]

where \( vᵢ \) and \( vᵢ \) are the sound velocities of the burst components due to the dispersion.

To verify the usefulness of this method test impact data have been analyzed. Fig. 7 shows three components of a burst, generated by a test impact on the hot leg of the main coolant piping of a FWR. The components are filtered with frequency bands of 2.5 kHz - 5 kHz, 8.25 kHz - 8.75 kHz and 5.25 kHz - 6.25 kHz, respectively. The measured times corresponding to the rising edges of the burst components are 9.2 ms, 9.2 ms and 10.0 ms, respectively. Taking the 8.5 kHz data as reference the determined time differences as function of frequency can be plotted (circles in Fig. 8).

These time differences fall on two lines which represent the dispersion of the a₉- and s₈-modes of the sound, respectively. For the thickness of the coolant piping of 50 mm, and the different frequencies of the burst components the theoretical sound velocities can be deduced from the lamb diagram of Fig. 6.
Fig. 7 Three components of a burst generated by a test impact on the hot leg of a PWR after bandpass filtering (a: 2.5 kHz - 5 kHz, b: 8.25 kHz - 8.75 kHz, c: 5.25 kHz - 6.26 kHz)

Fig. 8 Display of the time differences $t_1 - t_2$ of the three burst components of Fig. 7 as function of frequency $f$
The distance from the accelerometer to the sound source can now be calculated by inserting the thus derived velocities and time lags into the above given formula.

We obtained \((D+L) = (4.75 \pm 0.6) \text{ m}\).

This distance of the sound source agrees well with the real sound path of 5.4 m.

A similar analysis has been performed for a test impact at a BWR-plant. The circles in Fig. 9 show the time differences of the bursts components. The \(s\)- and \(a\)-sound modes are clearly separated. The dotted lines are the theoretical dispersion curves for a sound source in a distance of 14 m. The real distance from the impact location to the accelerometer was 13.2 m. These examples show that impact locations can be determined with an uncertainty of only about 10 %.

![Fig. 9 Display of the time differences \(t - t\) as function of frequency for a test impact at the reactor pressure vessel of a BWR plant](image)

This impact localization method has also been applied to cases of real bursts from loose parts alarms:

In a PWR very high and steep bursts led to alarms from the LPMS (Fig. 10). From the time differences between the bursts in different monitoring channels the region of the reactor pressure vessel could be inferred, where the sound anomalies reach the outer surface of the pressure vessel. Due to this estimate and because of the steep rising edges of the bursts one would have been concluded that the sound anomalies originated from the pressure vessel flange. Using the above described method of sound mode separation however we deduced the length of the sound path from the source to the accelerometer to be 3.4 m (Fig. 11). This indicated the sound source to be is the upper region of the core. Inspections of this region during the refuelling outage revealed a broken centre bolt in the upper core structure. It is not certain however whether the bursts were generated by the broken bolt which could move in the upper part of the fuel element or whether the fuel element head impacted against the core baffle.
Fig. 10 Examples of a burst signal from the reactor pressure vessel as registered in different monitoring channels.

Fig. 11 Time differences between filtered burst components as function of frequency. The lines are theoretical values for a wall thickness of 80 mm (core barrel) and a sound path of 3.4 m.
4. CONCLUSION
The examples given demonstrate the whole spectrum of application and the advantages of the LPMS:

- improvement of operation of the plant (valve hammering)
- improvement of availability (detection of loosened bolt before damage could occur)
- avoidance of further damage (easy securement of detached part)
- reduction of radiation exposure to service personnel

The new burst analysis method of sound mode separation as applied in the last example, permits the localization of the source of the sound anomaly with fewer, if necessary only one, accelerometer signals. The region where the sound reaches the outer surface of the reactor pressure vessel can be derived by the normal triangulation method, while the exact distance to the sound source is inferred by use of the new method. The new method also gives more reliable results because the dispersion of the sound velocities is properly taken into account. This also implies, that derivation of sound velocities from test impacts are superfluous.

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ADVANCED BURST PROCESSING METHODS IN LOOSE PARTS MONITORING

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ABSTRACT

Acoustic monitoring for detection of loosened or detached parts is recognized as an important technique for incipient failure detection at primary system of nuclear power plants. Appropriate hardware systems are installed at all German plants. The precision of the analysis statements, however, is essentially dependent on the used diagnosis methods. Increased application of computer based interactive systems for the analysis of acoustic burst signals can enhance the security of results considerably. Thus specific analysis tools have been developed comprehending the following items:

- Verification of burst impact events in acoustic signals,
- time analysis of series of impact events,
- plant-specific source location,
- application of Hertz theory for determination of energy and mass of impacting part or component.

The corresponding methods are now tentatively used in the frame of measurement analyses. Experiences with this methods and results will be presented.

KEYWORDS

Noise analysis; loose parts monitoring; acoustic monitoring; PWR; BWR; burst; arrival time; source location; Hertz theory; mass estimation; burst data base.

INTRODUCTION

Loose parts monitoring (LPM) is recognized as an important method for early failure detection at primary system of light water reactors. Appropriate hardware systems are installed in all German Nuclear Power Plants. Diagnostic results concerning the integrity of primary system and internals can be achieved on the basis of the interpretation of identified acoustic signals. In the past in a series of cases important prognosis could be reached, successful use has been reported from several countries /1-5/.

A determination of status of actual experience confirms the basic conception of acoustic monitoring. It shows on the other hand the following issues: deciding for the common acceptance of loose parts monitoring by authorities and utilities is the accuracy and reliability of diagnosis results. Thus high precision for the performance of measurement and analyses is asked as well as a well-founded knowledge for the interpretation of signals and their deviations. A common knowledge basis for important series of possible mechanical impacts at reactor vessel, steam generator, main coolant pumps and their internals is not sufficiently available till now. Important progress can be achieved by provision of extended catalogues of analysed events of operationally determined kind, or failure determined kind, or failure precursive kind. Plant specific source location codes are neces-
mary. Therefore current activities concentrate on increased use of computer-based dialogue systems in combination with interpreted acoustic signals to increase the reliability of loose parts monitoring as well as the precision of prognosis. The basic work for this objects at GRS has been sponsored by the Bundesministerium für Umwelt, Naturschutz und Reaktorsicherheit BMU (Federal Ministry for the Environment, Nature Conservation and Reactor Safety), which is highly acknowledged.

ACOUSTIC MONITORING OF PRIMARY SYSTEM

Detection capability of acoustic monitoring

Basic idea of loose parts monitoring is to survey the operational acoustic noise at the primary system by a suitable accelerometer instrumentation. Monitoring task is the detection of possible mechanical impacts; they can be identified by bursts in the acoustic signals. Fig. 1 shows the arrangement of the instrumentation with 14 accelerometers for a 4-loop-PWR and 8 accelerometers for a BWR (1300 MWe); the sensor positions are marked with small circles. Our experience is that mechanical impacts will be detected if they are generated by loose parts with masses of more than 10 grams impacting at natural collection areas like the lower plenum of reactor vessel or the inlet chamber of a steam generator.

![Sensor](image)

**Fig. 1:** Sensor positions of loose parts monitoring system

**Fig. 2:** Detection capability of acoustic monitoring

The detection capability of the sensor system for metallic impacts or contacts is high. This is due to the weak attenuation of structure borne sound in metallic structures, the high sensitivity of the sensors and a decrease of background intensity with frequency.

Some examples characterizing the detection capabilities are shown in fig. 2; they are documented by analog oscillograms with a time scale of 40 ms for all 6 signals (sensor positions are mentioned in fig. 2); the amplitudes are differently scaled for illustrative purposes. Three kinds of impacts with different origin are presented. In the left column burst signals are shown generated by loose parts in natural collection areas: the upper oscillogram corresponds to an impact in the lower plenum of reactor vessel of a PWR, the lower one to an impact in the inlet plenum of a steam generator. Short rise time, high signal to noise ratio (s/n ratio) and simple shape of the bursts are characteristic for this type of loose parts. In the medium column two examples of loosened parts are listed: the upper oscillogram shows a burst generated by a loosened part of the mounting of a primary coolant pump, measured by a sensor at the steam generator; the lower one shows a burst caused by a one-sided broken connecting link within steam separator cyclone tubes of a BWR. The right column presents burst signals of impacts which are not generated within the primary system but at neighbouring areas. The upper oscillogram shows a burst generated by a shaft contact with the casing of a feedwater check valve, the lower one by a spring contact within an hanger of heat removal system. The last examples are characterized by typical long periods, complex shape and specific frequency contents of the burst. Comprehensively it can be stated, that the detection capabilities are sufficient for the task of monitoring important active and passive components at primary system.
Trend to condition monitoring

The necessity to detect unexpected component failures as early as possible for safety reasons as well as the high costs of unplanned shut-down of NPPs constitute a trend to condition monitoring of active and passive components. Acoustic monitoring of primary system by means of LPMS can indicate mechanical contacts and deviations of the following components:

- control rod guide tubes
- fuel assemblies
- core barrel
- core support structure
- steam generator tube bundles
- coolant pumps
- check valves

Periodical acoustic measurements - as they are required by the German guidelines, 1/6/ - can therefore provide a valuable tool in the framework of condition based maintenance. Our experience is that this trend is continued with increasing plant age and by enhanced use of digital analysis systems.

INTERACTIVE ANALYSIS OF BURST ENSEMBLES

Acoustic analysis system at GRS

Whereas in the past analog burst processing concentrated primarily on single events, handling of extended burst pattern ensembles with typically 50 or 100 events is enabled by newer developments on the basis of digital burst processing. Experience of GRS with measurement campaigns have lead to the conception of a fast digital burst data processing system with a configuration, that has as major components a multi-channel transient recorder, a computer work station and a fast Winchester disk storage. Our MEDEA system for off-line analysis is used for burst data capture, data analysis and data storage. Data inputs can be provided by FM tapes or an IBM PC or from a 4-channel transient recorder (see fig. 3). The system is installed on a HP 32 bit workstation.

![Diagram of acoustic analysis system]

- LPM measurements of 10 NPP (8 PWR, 4 BWR)
- LPM analog signals stored on 400 FM-tapes
- 3500 digitized burst signals (digitally stored)
- up to 10 channel parallel, typically 4k samples/burst, digitizing rate 100 kHz
- processed by 32 bit workstation
- interactive analysis software, colour graphics
- data base system DB002

Fig. 3: MEDEA-System for off-line acoustic monitoring analysis

Interactive burst analysis software has been developed and tested in our laboratory. Graphic display of burst pattern with optimum amplitude scaling is possible in parallel of 1 to 10 channels (see fig. 5); time windowing (zooming) is interactively enabled by a cursor or by digital input. Based on the original digitized data points the burst envelope and the burst rms-signal can be displayed. Calculation of envelope is performed by searching and displaying for the local maximum of a period of five neighboring points, which is shifted sequentially over the whole time window. Rms-signal is calculated with an interactively selected integration time constant.

The empirical experience of GRS in acoustic monitoring of primary systems includes till now data of measurements and analyses of ten different German nuclear power plants (fig. 4). To coordinate the experience and to make it available in a more general way a central burst data base is used which is discussed later on. 3500 burst signals are available. Typical digitizing rates are 100 KHz, the time windows are usually 40 ms with up to 160 ms for each burst including its pre-history.
Burst verification

If burst s/n ratios are below 3:1, identification and verification can be a problem for digitally processed burst signals. For distinguishing between background and burst signal we use within the time windows of 40 ms with typical 30% pre-trigger three test areas of each 1 ms period. These are: a reference area at the beginning of the time window for background calculation, a burst test area just after the trigger point and a spike test area, which begins 1 ms after the trigger point. Burst propagation times are taken in consideration. RMS values of the test areas are calculated for each burst pattern. A signal which has triggered is verified as a burst, if RMS value within the burst test area is at least three times of that of background area (and it is no spike according to spike testing). Our experience shows that spikes and drop-outs (of FM tapes) are detected with a reasonable efficiency.

Arrival time determination

Burst arrival times are deciding for source location and for all burst parameter calculations. As the burst arrival can be hidden in the signal background, they have to be determined carefully. According to wave mode dispersion, the subjective experience of the expert is important.

A zero-crossing high resolution display used as an expert tool is shown in fig. 5 for 2 channels. As arrival time coincides with a zero-crossing of the burst signal, all signal zero-crossings in an interactively selected interval at the computer display are marked by vertical lines on a high resolution pattern of the burst. The selected arrival time is individually displayed in digital form for further use.

A new algorithm has been developed for automatic arrival time determination which uses the burst frequency information, zero-crossing and envelope. It has been tested at an ensemble of 56 complex and individually shaped bursts, all with a s/n ratio > 3:1. According to an expert valuation the arrival times have been correctly determined with more than 90 % certainty.

Fig. 5: Interactive arrival time determination by zero-crossing display

Fig. 6: Group velocities, burst arrival time automatically calculated

Fig. 6 shows in the lower part a plot of algorithmically determined arrival times of 78 bursts (13 test impacts, 6 sensors) as a function of the known wave path. The slope of the interpolated line in fig. 6 yields an experimental group velocity of 5.1 m/ms; this value is in good agreement (upper part of fig. 6) with the theoretical values of s-group velocities of plate waves for the dominant frequencies 6-10 kHz, calculated for a wall-thickness of 139 mm and confirms the correctness of the used algorithm.
Interval time analysis

An identification of an impacting component of reactor or steam generator internals can be reached, if the analysis of subsequent bursts yields time intervals, which show dominant frequencies, that can be associated with known vibrator frequencies. An example of an successful identification is shown in fig. 7.

![Diagram showing frequency distribution of burst interval time differences](image)

Fig. 7: Frequency distribution of burst interval time differences

Measured by a sensor at PWR reactor vessel head, 198 burst time intervals have been analysed. Converted to frequency range with a frequency resolution of 1 Hz, the plot of the frequency distribution of interval time differences (fig. 7) showed distinct maxima at 2, 6, 10, 14 Hz. These could be associated with 1. and 2. beam mode vibration frequencies of fuel assemblies, which are eigenfrequencies known from vibration monitoring analysis: mechanical contacts of fuel assemblies could therefore be diagnosed. This interpretation has been confirmed by the plant inspection; it showed imprints at several fuel assembly heads.

Mass estimation

Contact times of impacts of loose parts, which are typically in the order of 30 to 300 µs, can be calculated by using the governing equations of Hertz-theory of impacts /8/. Adversely these equations can be used for the estimation of the mass of an impacting loose part. First results have been published in /1/.

According to the great number of the determining parameters of contact time, signal analysis of burst in time domain for this purpose seems fruitful only in limited cases. Due to the fact that the nearly half-sine shape of contact force governs the impact process, studies have been concentrated to frequency representation of impacts. In the following the method shall be illustrated on a practical example.

A typical burst corresponding to an impact of a loose part (m=43g) in the hot chamber of a steam generator is shown in upper left part of fig. 8. The right part of the figure shows computed power spectral densities (PSD) of the burst, starting from its arrival time and the background (first 20.48 ms of the background) in the lower and upper part, respectively in the 50 kHz frequency range. Whereas PSD of background shows only relevant frequency parts at magnet adaption resonance frequency of the sensor (6 kHz) according to the lower frequency contents of operational acoustic noise of the fluid, PSD of the burst shows additional frequency parts which reach in the higher frequency range and excite sensor crystal resonance frequency at 21 kHz; a specific decrease of their intensity with frequency can be stated. Result of calculation by Hertz-theory of the PSD of contact force for a contact time of 44 µs corresponding to the loose part impact is shown in left lower part of fig. 8. Slope of this curve is nearly identical to the intensity decrease of the burst PSD and confirms that burst PSD can be used as a quantity for mass estimation. The studies have still not reached their final stage, thus further results are to be expected.
Source location

If at a plant bursts are detected, questions for the location and origin of the primary events have to be answered. As the acoustic generation and propagation characteristics of structure-borne sound within the primary system are very complex, source location algorithms are preferable which can be interactively processed on the graphic computer display. For illustrative purposes in the following a hyperbola intersection method is used; direct location /7/ and circle intersection method /1/ yield identical results. For source location by arrival times we stored a set of calculated group propagation velocities of the relevant plate waves - which are depending from wall-thickness of the component and the frequency band - as well as a schematic vertical projection and development of major components (reactor vessel, steam generator, fig. 10, 12, 16) as a graphic in the computer. Results of source location can then interactively be displayed on the shown development of the component.

Verification of the source location algorithms have been performed with burst arrival times, deduced from test impacts with known impact position. First example shows in fig. 9 the burst pattern of an test impact at the motor flange of an internal coolant pump of a BWR; the selected burst arrival times are marked. Source location by intersection of hyperbola yielded - as photographed from schematic graphic display of structure - the correct location result (fig. 10): the specific reactor coolant pump.

A further example is given in fig. 11 and 12 for a PWR steam generator. Fig. 11 shows the burst pattern with the burst arrival times marked of an test impact at a tube support grid, performed with an metallic rod during plant shut-down. Fig. 12 presents the schematic graphic display of the calculated result of the source location. Location result and real impact position are in good accordance.
BURST DATA BASE PROCESSING

The empirical experience of the interpretation of acoustic burst signals reached so far, is of specific relevance for future diagnosis and prediction. As the understanding of bursts of same basic phenomenon can be of great help for interpretation of unknown events a fast access possibility for signal patterns of known and analyzed details became necessary. The setup of a central burst data base with name D9002 has been established. Fast access to the data, interactive display with flexible graphics and problem-oriented keywords have been some of the requirements which could be fulfilled. Actual access times are now in order of up to some seconds and are acceptable.

Network structure of data base

Aim of a suitable classification of burst signals is to extend the diagnosis possibilities by guided access to interpreted stored signal patterns. Reference characterization of impact events is performed according to the scheme in fig. 13: impact type, mass and energy of impactor and wave path from impact position to sensor, separated for BWR and PWR. In an actual case a relevant subset can be selected.

- Impact type - PWR/BWR
  - loose part, impact position
  - loosened part, impact position
  - neighbouring area impacts, impact position
  - test impact, impact position
  - reference measurement

- mass of impactor (grams)
- energy of impactor (Nm)
- wave path from impact position to sensor (m)

Fig.13: Data base characterization of impact events

Reasonable plant-specific and event-oriented data access is guaranteed by the database structure (fig. 14). The data are organized in a network structure with two different kinds of data sets: automatic master data sets (marked with "A") and detail data sets (marked by "D"). They are linked by data paths. The managing of the measurement data is performed in a leading header data set "Header" which contains a data catalog with information about each measurement with the items: plant, measurement, header number, measurement date, analog tape, digital tape, info, function and event. The capacity of this detail data set has been preset to 250 measurement entries; an extension is possible. Second detail data set "Next header" contains the next free header number. 9 additional automatic master data sets are containing the key items which provide an easy way to establish multiple keyed access to detail data; they are automatically updated for each new measurement input. Detail data set "Result" contains calculated numerical burst quantities which are stored in the data item field "Result (1:20)".

Fig.14: Scheme of burst data base
The following characteristic values are believed to be important for analysis and interpretation of burst ensembles:

- times
  - burst arrival time
  - time of the first local maximum
  - time of burst maximum
  - risetime
  - period

- amplitudes
  - first local maximum
  - burst maximum
  - background
  - s/n ratio
  - rms

- diverse
  - rise
  - dominant frequency
  - quality factor
  - burst form class

- global parameters
  - intensity
  - area
  - fine structure

- external parameters
  - wave path
  - mass of impactor
  - energy of impactor

\[ y(t) = \int_{-\infty}^{t} x^2(u) \, du \]

- \[ R_m = \text{Max} \{ y(t) \} \]
- \[ I_m = \frac{1}{t_b - t_a} \int_{t_a}^{t_b} y^2(t) \, dt \]
- \[ A_r = \frac{1}{t_b - t_a} \int_{t_a}^{t_b} y(t) \, dt \]
- \[ F_r = \frac{1}{t_b - t_a} \int_{t_a}^{t_b} \dot{y}(t) \, dt \]

\( x(t) = \text{time signal} \quad t_a = \text{burst arrival time} \quad t_b = \text{burst analysis time} \quad t_R = \text{rms-integration time} \quad y(t) = \text{rms-signal} \)

Fig. 15: Shape determined parameters of bursts

Fig. 16: Characteristic burst parameters

Shape determined global parameters of single bursts: burst intensity, burst area and burst fine structure are calculated from the rms-signal of the burst (fig. 15). As the result of the rms-transformation is dependent of the rms-integration time-constant \( t_R \), our experience is to use \( t_R = 1 \) ms, which keeps the contour of the burst with sufficient degree. Global parameters are normalized by dividing with Rms-max; they are calculated for a burst analysis time of 10 ms, starting with burst arrival time. Selected results of a burst parameter calculation are shown for the burst in fig. 16. Comparable results are available for all stored bursts in numerical burst database D 9002.

**TYPICAL ANALYSIS EXAMPLES**

Reference analyses of bursts of the same basic phenomenon are of great help for the interpretation of unknown burst events. Varying impact locations and impact velocities of a loose part in a noisy background lead to characteristic burst quantities which vary within a corresponding range. Measurement of the similarity or dissimilarity of different burst ensembles by use of the previous discussed burst parameters can be a valuable tool for determining the basic sound generation and propagation relations. The following examples are based on automatically determined burst arrival times.
A separation of burst ensembles is presented in fig. 17 for different acoustic wave paths (6 sensors). For a series of test impacts at a BWR vessel upper flange burst parameters have been automatically computed for the 79 studied bursts. Whereas the single parameters burst maximum amplitude and burst area wouldn't separate the different burst ensembles (represented by the axes in fig. 17) the 2-parameter correlation of these quantities shows 6 distinct clusters according to 6 different wave paths. The data confirm that there is a clear contour separation for the individual burst ensembles. The 2-parameter-correlation can therefore be used for wave path estimation.

A second example of processing of burst ensembles is given in fig. 18: Based on automatically computed arrival times of 13 2-channel burst patterns - measured during 100 % plant operation - the frequency distribution of arrival time differences is shown, as a display photograph. The arrival time differences show a distinct sharp band with a maximum at + 0.2 ms. This shape is typical for loose-ned parts impacts and corresponds well to the observed fact, that a source at a fixed position generated the studied bursts.

Result of rise time calculations of a burst ensemble which corresponds to impacts of a loose part in steam generator inlet plenum are presented in fig. 19. The frequency distribution of rise times is plotted. A distinct maximum with a value 0.23 ms can clearly be identified. Nearly all rise times are less than 1 ms. This is in good accordance with our empirical experience: for impacts with acoustical wave paths of less than 3 m as being the case in this example a rise time of less than 1 ms can be stated.
In fig. 20 is shown, how burst patterns, measured during start-up phase of a BWR, can be evaluated on the basis of a source location. The result of hyperbola intersection source location - one of 4 feedwater pipes, see fig. 20 - on basis of burst arrival times, permitted to associate the measured burst patterns with shaft contacts of a medium regulated feedwater check valve; these are known to be dependent from actual shaft position within the bearing. This evaluation was confirmed by the consequent 100%-plant operation at which no corresponding bursts have been observed. Major consequences for plant operation therefore had not to be undertaken.

The example in fig. 21 shows, how inspection findings and source location result will coincide. Inspection of upper part of a steam generator during plant shut-down showed that the end of a spacer grid at a certain position could uncritically touch inner steam generator shroud. During 100%-plant operation burst patterns have been measured. Source location by hyperbola intersection yielded just this point as the result (see fig. 21). Therefore no major plant consequence was necessary.

CONCLUSIONS

The recently shown examples make evident that from view of early failure detection the methodic of acoustic monitoring cover important kinds and areas of mechanical contacts at primary system. Increasing plant age and enhanced use of digital analysis systems will intensify the trend to condition monitoring of actual component status. Precision and confidence of analysis statements will be enhanced furthermore by the technique of processing digitized burst ensembles - with a corresponding increase of statistic security - and by use of location algorithms, which can be interactively processed on the graphic computer display. Verification of Hertz-theory models for mass-estimation purposes are under work, further improvements are expected.

When at a plant unexpected bursts are detected monitoring systems should be able to perform data capture and first analysis. Access to a specialized knowledge basis should be possible, which can be located for example at headquarters of major utilities. GRS is preparing for specific purposes a data line to its data base which contains an increasing number of reference patterns. Fast, event-oriented and plant-specific access has been established, further extensions to a common knowledge bases are expected.
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WATER LEAK DETECTION IN STEAM GENERATOR OF SUPER PHENIX

M. BRUNET, P. GARNAUD, D. GHALEB* and N. KONG†

CEA Cadarache,
CEA Saclay and
EDF Chatou, France

Abstract - With the intent of detecting water leaks inside steam generators, we developed a third system, called acoustic detector, to complement hydrogen detectors and rupture disks (burst disks). The role of the acoustic system is to enable rapid intervention in the event of a leak growing rapidly which could rupture neighbouring tubes. In such a case, the detectable flow rate of the leak varies from a few tens of g/s to a few hundred g/s.

At the SUPER PHENIX, three teams work in parallel in complementary frequency bands: EDF [0-20 kHz], CEA/SPCI* [20-100 kHz] and CEA/STA† [50-300 kHz]. The simulation of water leaks in the steam generator by the argon injections performed to date at 50% of the rated power has shown promising results. An anomaly in the evolution of the background noise at more than 50% loading of one of the two instrumented steam generators would make difficult any extrapolation to full power behaviour.

1. INTRODUCTION

The operating safety and structural integrity of steam generators (SG) in the fast reactor field depend on early detection of water-sodium reactions to allow control of secondary damages related to erosion-corrosion and wastage phenomena [1]. To this end, fast reactor controls, and those of the SUPER PHENIX in particular, are equipped by design with two complementary devices: hydrogen detectors and rupture disk devices. Hydrogen detectors [2] are intended to detect the appearance of a small leak by the hydrogen produced in the reaction, while rupture disk devices serve to dry up the steam generator when the overpressure from the reaction causes them to rupture.

Obviously, the effective ranges of these two devices are very different. Whereas the hydrogen detector is basically able to detect very small leaks on the order of one gram of water per second, with a response time of one minute, the rupture disk responds in a few milliseconds to the pressure wave generated by leaks of about one kilogram per second. Between these two leakage ranges—from 10 g/s to 1 kg/s—there is a range of flows too small to rupture a safety diaphragm, but large enough to cause, within the response time of the hydrogen detection system, damages to the internal structures of the steam generator.

This fact prompted us to investigate a third monitoring alternative, having an intermediate sensitivity and a quick response time, both of which characteristics are compatible with the capabilities of acoustic monitoring devices [3]. Electricité de France (EDF) and the French atomic energy commission, Commissariat à l’Energie Atomique (CEA), are jointly involved in research on this subject at the SUPER PHENIX reactor, which has already shown promising results.

2. PRINCIPLES OF ACOUSTIC DETECTION

A water circuit tightness failure within a steam generator causes water or steam to be injected under about 180 bar of pressure into the volume of sodium around it. Such an injection produces the following secondary effects:

- vibration of the supporting structures due to the sonic outflow through the orifice;

- generation of pressure waves in the sodium by primary bubbling (water) and secondary bubbling (gaseous products of the reaction, in particular hydrogen), and due to the turbulence of the jet;

* SPCI and STA are Atomic Energy Commission laboratories.
- and emission of stress waves due to the rapid increase in the temperature of the surrounding structures.

These three phenomena all generate waves which we term acoustic, that can propagate in the generator space either via the metal structures, the sodium or more likely via an exchange of mechanical energy between the two mediums. These signals finally appear on the external shell of the steam generator where suitable transducers convert them to electrical signals.

As the acoustic signals emitted by the leak are not correlated with other sound sources normally occurring in the secondary circuit, their detection consists in comparing an assumed steady-state, leak-free reference condition where only background noise is present with the same condition having additionally a leakage noise.

A fundamental feature of all these events is the wide emission band, ranging from a few hertz to a several hundred kilohertz [4]. This is why three frequency bands are studied in parallel, as follows: 0–20 kHz (EDF), 20–100 kHz (CEA/SPCI) and 50–300 kHz (CEA/STA) to determine whether the detection has an optimum sensitivity in one of these frequency bands of the overall emissive frequency range.

The experimental qualification of these detector instruments, presented here, requires a comparison of signals with and without a leak. Since it is inconceivable to deliberately produce a water or steam leak in the SUPER PHENIX steam generator, we approached the problem by postulating the separability of the two effects:

- we studied, in the actual generator, the acoustic signal due to reactor operation (background noise) and the signal altered by a simulated leak (injection of an inert gas);
- we compared, in a experimental loop, the signal transmitted by the simulated leak with that produced by a real leak into the sodium.

3. DESCRIPTION OF DETECTION EQUIPMENT

3.1. Measurement Principle

Acoustic signals present on the SG shell are detected by piezoelectric transducers fitted to the cold end of a waveguide (a cylindrical, stainless stell rod) welded to the shell.

The frequency response of the sensors used by EDF and CEA/SPCI can be considered linear and that of the sensors used by CEA/STA exhibits a significant resonance around 250 kHz.

Only two of the four SUPER PHENIX steam generators are instrumented, as follows:

- 32 sensors serving EDF and the CEA distributed over 8 localization levels on SG D (Fig. 1);
- 4 EDF sensors distributed about SG C.

The signals delivered by the sensors are conditioned and processed differently according to the frequency bands used.

3.2. CEA Configurations

The low-level electrical signals from the sensors are preamplified near the sensors and then amplified, shaped and filtered in the control room.

*CEA/SPCI (20–100 kHz)

The signal thus conditioned is sent to the processing system which calculates, on the one hand, the RMS value and on the other hand the power spectral density. Detection of an anomaly occurs either when the last-computed RMS value exceeds a suitable threshold or through pattern recognition and comparison of frequency spectrums. In both cases, operator alarms are enabled or disabled on the basis of trends in SG operating parameters.

*CEA/STA (50–300 kHz)

The processing provided by this setup is limited to computation of the RMS values of the acoustic signals, followed by treatment of overshoot of preset thresholds. Alarm trip thresholds are adjusted for different background noise levels and steam generator operating parameters to avoid false alarms.
3.3. EDF Configuration

The signals from each accelerometer are conditioned by charge amplifiers, then transmitted to a processing system.

First, spectral analysis is performed on 100 equal-width frequency bands (between 0 and 20 kHz) of the signals. A real-time algorithm evaluates the sound power trend in specific, frequency bands determined beforehand by Primary Component Analysis. Alarm generation also takes into account certain rules of consistency between channels and between steam generators.

4. QUALIFICATION TESTS

For purposes of determining alarm threshold values and choosing the relevant frequency bands for the EDF processing, qualification tests at SUPER PHENIX were programmed in two stages: a first stage of isothermal tests and a second stage of tests under power. In each stage, the instrument's background noise values are measured and a series of argon injections are made to simulate water leaks into the sodium.

Various preliminary studies of experimental sodium loops showed an acceptable similarity between the noise produced by a steam leak into the sodium and the noise produced by the injection of inert gas through a small orifice. Figure 2 for example shows similar power spectral densities for argon and water injections at the same mass flow rate in sodium at 500°C, through an orifice 0.8 mm in diameter.

The high pressure argon injection device, installed on SG D (Fig. 1, 8-202° plan) is provided with a 2 mm diameter injection orifice and can operate at pressures between 40 and 140 bar.
5. RESULTS OF QUALIFICATION TESTS

Measurements are to be provided for three SG operating situations: isothermal (no load) and at 50% and 100% of rated power.

In each situation a series of injections of argon at pressures between 40 and 140 bar are planned through the injection orifice at the bottom of S.G.

To this date, only the first two series of injections have been possible.

5.1. Isothermal Results

Under isothermal conditions, the main source of noise in the intermediate loop is composed of noises from the flow of sodium determined by the rotational speed range of the secondary pump; this noise increases with pump speed [5]. The amplitude of the background noise generally increases exponentially as the square of the pump speed for frequencies under 100 kHz and monotonically at higher frequencies (50-300 kHz).

In this situation, where no boiling noise is present, the signal-to-noise ratio recorded during the different injections is particularly favorable, irrespective of the frequency bands investigated (Table 1 - Figure 3).

In the double phase medium we noted an attenuation of the low frequency noise (< 7 kHz). Nevertheless, the presence of gas bubbles is better felt at high frequencies.

5.2. Results at 50% of Rated Power

The measurements initially planned for 50% rated load operation were in reality made between 40% and 48% of rated power.

These tests allow us to make the following observations:

- the noise level was substantially the same at 50% of rated load as in nominal isothermal (no load) operation for frequencies between 100 and 300 kHz. However, at frequencies below 100 kHz, noise amplitude at 50% loading is multiplied by 4 on the average, compared with no
load operation;
- all the injections made were detected by the set of sensors (Fig. 4) and S/N ratios were very good, as shown in Table 1;

Figure 3: ARGON INJECTION IN ISOTHERMAL CONDITION

Figure 4: ACOUSTIC SIGNALS OF LEVEL 2 AND 8 DURING AN ARGON INJECTION (140 BAR) AT 50% POWER
- in a certain way, the strength of the signal picked up depends on the distance between the source and the sensor. The amplitude of the signal from the sensor in plane 8—nearest the injector—is greater than the signal from the sensor in plane 2 (Fig. 4).

5.3. Extrapolation to Full Power

Pending accomplishment of the injection tests at full-power, it was interesting to make a prediction based upon the available information. Data is indeed available on the background noise trend up to 100% of nominal load for frequencies less than 100 kHz.

At less than 50% of rated power, the background noise level increases regularly with reactor power. Beyond that point there is a substantial and inconsistent increase in the noise trend of SG D (Fig. 5). The signals however show that this phenomenon does not appear in the 200 to 1000 Hz frequency range (Fig. 6). Moreover, this anomaly was not observed on SG C, at least at low frequencies. We are currently researching the origin of this aberration.

In the band of frequencies where the background noise grows at a regular rate, comparison with argon injections indicates that detection would be possible at nominal load.

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**Figure 5: Background Noise Versus Steam Generator Power**

[200–20000 Hz analysis]
Acoustic monitoring of the SUPER PHENIX steam generators under satisfactory conditions appears to be possible, as the tests made under conditions of normal SG behaviour indicate. An unexpected noise source appeared during build-up to full power, but the first analyses, and in particular the fact that this phenomenon is unique to SG D, indicate that correction is possible.

When the level of acoustic signals from a leak is sufficient, it should be possible to detect them all over the structure and multiplication of the sensors provides suitable redundancy of information. Also, the observed weakening of the signal-to-noise ratio with increased source-sensor distances allows us to consider the possibility of axial localization of leaks.

The three detection methods discussed do not as yet present any determining differences in performance and the final choice amongst them is likely to be made on the basis of the long term stability of the reference noise.

REFERENCES


<table>
<thead>
<tr>
<th>FREQUENCY RANGE</th>
<th>EDF [0-20 kHz]</th>
<th>CEA/SPCI [20-100 kHz]</th>
<th>CEA/STA [50-300 kHz]</th>
</tr>
</thead>
<tbody>
<tr>
<td></td>
<td>min.</td>
<td>max.</td>
<td>min.</td>
</tr>
<tr>
<td>Injections</td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>(bar)</td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>40</td>
<td>11(2/135°)</td>
<td>19(8/157°)</td>
<td>8(2/300°)</td>
</tr>
<tr>
<td>70</td>
<td>9(2/135°)</td>
<td>16(8/157°)</td>
<td>6(2/300°)</td>
</tr>
<tr>
<td>100</td>
<td>14(3/ 45°)</td>
<td>22(8/157°)</td>
<td>7(2/300°)</td>
</tr>
<tr>
<td>140</td>
<td>11(2/135°)</td>
<td>19(8/157°)</td>
<td>6(2/ 90°)</td>
</tr>
<tr>
<td>Injections</td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>(bar)</td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>50 % OF RATED POWER OPERATION</td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>40</td>
<td>3(2/315°)</td>
<td>9(8/157°)</td>
<td>6(2/225°)</td>
</tr>
<tr>
<td>70</td>
<td>6(2/315°)</td>
<td>14(8/157°)</td>
<td>8(3/ 45°)</td>
</tr>
<tr>
<td>100</td>
<td>6(2/135°)</td>
<td>12(8/157°)</td>
<td>6(4/225°)</td>
</tr>
<tr>
<td>140</td>
<td>6(4/315°)</td>
<td>13(8/157°)</td>
<td>8(2/225°)</td>
</tr>
</tbody>
</table>

11 (2/135°) 19 (8/157°) means: minimum S/N ratio is 11 dB measured on sensor placed in level 2 at 135° of angular position and maximum S/N ratio is 19 dB on sensor placed in level 8 at 157°.

**TABLE 1** SIGNAL TO NOISE RATIO VALUES (dB) MEASURED IN ISOTHERMAL CONDITION AND AT 50 % OF RATED POWER.
FLOW MEASUREMENTS/BOILING

Session chairman: A. Federico (Italy)
SUMMARY OF SESSION

The four papers presented in this session are evenly divided between in-core applications and out-of-pile simulations, with the objective of detecting subcooled boiling in PWRs or of characterizing two-phase flow patterns in BWR channels.

Defloor and Baeyens report on an anomaly which occurred at Doel 2 (PWR) in 1986. Unusually large values of the axial offset and of the readings from two thermocouples were observed, that could only be justified by the onset of nucleate boiling - presumably due to crud deposits induced by oxidized cladding - in a limited number of fuel pins. The analysis of the APSD of properly re-positioned fission chambers shows the typical signature of nucleate boiling in the upper part of the suspected pins. Even though some aspects of the phenomena are not fully understood, it is emphasized that neutron noise analysis supplies a convincing assessment of the anomaly, which is quantitatively supported by complementary calculations and measurements of reactivity effects in the core.

Per et al. describe analytical and experimental efforts spent at Paks 2 (PWR), aimed at extracting information on the presence of voids - and not only on their propagation velocities - from neutron noise measurements, by gaining insight into the meaning of the shape of the CPSD phase between axially placed neutron detectors. In fact, the interpretation of Paks 2 experiments and the supporting calculations confirm that a linear dependence of CPSD phase on frequency is a reliable indicator of the presence of hot spots in fuel pins - and therefore of the possible occurrence of local boiling. In particular, such linear dependence is found to be strongly correlated to the power of the pertinent pins and to over-heating of their surfaces; thus, its puzzling disappearances and re-appearances can be attributed to normal changes in the power form factor occurring in the course of the core fuel cycle.

Van der Hagen and Van der Voet illustrate the experimental campaign, carried out in an air-water loop simulating a BWR channel, aimed at measuring two-phase flow velocities via noise correlation techniques in a variety of flow regimes. The experimental setup is characterized by high simulation level: in fact, also neutrons and detectors are "modeled" by visible light of comparable relaxation length and by phototransistors, respectively. The velocities are accurately measured in the loop and significant conclusions are reached from the comparisons with the calculated values for both bubbly and slug flows. These types of analysis, however, are still precluded in real BWR channels, where the indispensable information about the steam distribution cannot be obtained experimentally.

Iida's paper is again related to the problem of local boiling detection in cases when temperature fluctuations are very small, as in PWRs and LMFRs. The author describes a developed and simplified version of a double-thermocouple instrument he presented at SMORN-IV. A single automatically compensated thermocouple is used in the present setup and temperature fluctuations are derived from the changes in the PSD of the (optimally) high- and low-pass filtered signal. In spite of the lower sensitivity exhibited by the 1-TC setup in mockup tests, it is concluded that it is of greater potential usefulness than the previous 2-TC system.
NUCLEATE BOILING DETECTED BY NEUTRON NOISE MONITORING AT DOEL 2

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ABSTRACT

The DOEL site in Belgium accommodates 4 nuclear power plants. DOEL 1 and DOEL 2 are pressurized water reactors with a thermal power of 1192 MWth. During the 11th operating cycle of DOEL 2, the core didn't behave as expected. The axial offset grew more negative than normal and two thermocouple readings increased.

Neutron noise measurements yielded a clear signature of nucleate boiling in certain fuel assemblies. The void fraction and the bubble speed were inferred from the noise spectra. The results agree with thermohydraulics and neutronics.

The affected assemblies all came from the same vendor and had high burnup. Inspection during refuelling revealed thick oxide layers on the fuel rods. We attribute the increased nucleate boiling rate to the abnormal surface condition.

1. INTRODUCTION

The DOEL site is located in Northern Belgium, about 15 km from Antwerp, on the left bank of the Scheldt river. There are 4 nuclear power plants with a total electrical output of 2810 MWe. All have pressurized water reactors.

DOEL 1 and DOEL 2 are identical and share many auxiliary circuits. The primary circuits have two loops. The main core parameters are:

<table>
<thead>
<tr>
<th>Parameter</th>
<th>Value</th>
</tr>
</thead>
<tbody>
<tr>
<td>Thermal power</td>
<td>1192 MWth</td>
</tr>
<tr>
<td>Primary pressure</td>
<td>155 bar</td>
</tr>
<tr>
<td>Average temperature</td>
<td>299.6 °C</td>
</tr>
<tr>
<td>Delta temperature</td>
<td>29.6 °C</td>
</tr>
<tr>
<td>Total number of fuel assemblies</td>
<td>121</td>
</tr>
<tr>
<td>Number of fuel rods in one assembly</td>
<td>179</td>
</tr>
<tr>
<td>Fuel rod diameter</td>
<td>1.07 cm</td>
</tr>
<tr>
<td>Active fuel length</td>
<td>244 cm (8 ft)</td>
</tr>
<tr>
<td>Average linear power density</td>
<td>226 Watt/cm</td>
</tr>
<tr>
<td>Number of core exit thermocouples</td>
<td>35</td>
</tr>
</tbody>
</table>

In the following section, we discuss core behaviour at DOEL 2 and we seek for a thermohydraulic explanation. We then switch to neutronics and neutron noise analysis, in order to substantiate our conclusions. After that we give an overall picture and we elaborate on the root causes.
2. CORE BEHAVIOUR

On October 5, 1985 the 11th reactor cycle of DOEL 2 started just like the preceding ones. During the spring of 1986 however, the core exhibited a behaviour which hadn't been observed before. The axial offset grew more negative than normal and two thermocouple readings slowly increased (Fig. 1 and 2).

The calibration of the thermocouples and the fission chambers was carefully checked. No anomalies were found. Apparently, something was going on inside the reactor.

The flux maps looked normal. So did the hot channel factor and the hot spot factor. No Technical Specification limit came close.

The fuel vendor made a preliminary safety evaluation. Nuclear safety didn't seem challenged. The DNB ratio remained acceptable. A surveillance program was implemented, nevertheless. The safety evaluation was refined as more information became available.

Core behaviour became more aberrant with time. The flux map and the thermocouple map of May 6, 1986 (Core burnup : 7812 MWD/Ton) gave the following picture:

<table>
<thead>
<tr>
<th>Core axial offset</th>
<th>328.8 °C</th>
<th>335.4 °C</th>
</tr>
</thead>
<tbody>
<tr>
<td>Thermocouple 18 (Position G4)</td>
<td>-6.4 %</td>
<td>-3.5 %</td>
</tr>
<tr>
<td>Thermocouple 24 (Position D7)</td>
<td>316.0 °C</td>
<td>316.0 °C</td>
</tr>
</tbody>
</table>

The positions G4 and D7 were occupied by 2 third cycle fuel assemblies. They belonged to the group of 12 assemblies that would reach the highest burnup : 37000 MWD/Ton without stretch out. The other 10 assemblies didn't bear any thermocouples, so exit temperatures are unknown. We believe, however, they were also affected, however. The power distribution in the core would be tilted otherwise.

Most of the fuel that was loaded for cycle 11, came from the same vendor. Some assemblies were visually examined after cycle 11. The oxide layer found on the fuel rods was thicker than expected. The thickness increased with the square of the burnup. The oxide layer looked brittle and subject to spallation. This left a surface with a rough appearance.

Fig. 3 shows the axial flux profile in the assembly at position G4 and in a normal assembly with comparable burnup. At G4 the flux in the upper half of the assembly is significantly lower. This also shows up in the (local) axial offset (Flux map of May 6, 1986):

<table>
<thead>
<tr>
<th>Core position</th>
<th>Oxide layer</th>
<th>Relative power</th>
<th>Axial offset</th>
</tr>
</thead>
<tbody>
<tr>
<td>G4</td>
<td>Thick</td>
<td>1.042</td>
<td>-8.7 %</td>
</tr>
<tr>
<td>D5</td>
<td>Normal</td>
<td>1.052</td>
<td>-4.9 %</td>
</tr>
<tr>
<td>D7</td>
<td>Thick</td>
<td>1.055</td>
<td>-9.9 %</td>
</tr>
</tbody>
</table>

The axial offset at G4 and D7 resembles the axial offset in an assembly carrying a control rod.

It's obvious to assume a link between the phenomena.

3. PRESSURE DROP

The higher thermocouple readings point to a decrease of the coolant flow. According to Fig. 3, only the upper half of the assembly seems affected. In the lower half, the neutron flux looks normal.

The DOEL 2 core has an open lattice and the cross flow resistance is small. As the flow takes the path of least resistance, any obstruction in a fuel assembly causes a flow redistribution.

Suppose the delta temperature in the upper half of the assembly is directly proportional with thermal power and inversely proportional with flow rate.
We then conclude from the thermocouple map of May 6, 1986:

<table>
<thead>
<tr>
<th>Core position</th>
<th>Flow rate</th>
</tr>
</thead>
<tbody>
<tr>
<td>G4</td>
<td>52%</td>
</tr>
<tr>
<td>D7</td>
<td>41%</td>
</tr>
</tbody>
</table>

(Normal flow rate = 100%)

To explain the thermocouple readings, we have to assume a 50% flow reduction in the upper half of the assemblies! As \( \Delta \rho \) is roughly proportional with the square of the flow rate, this implies a fourfold increase of the pressure drop coefficient.

Several explanations are conceivable:

a) The spacer grids were partially plugged with crud from the cladding. This is unlikely:

- The flow speed in the core is high (3.8 m/s). If spallation occurred, the crud would be very fine.
- The crud concentration in the primary circuit remained normal during cycle II.
- A visual examination of the grids during refuelling didn't reveal any anomaly.

b) The surface roughness of the fuel rods had increased. This could be a factor: At reactor conditions (Reynolds number = \( 4 \times 10^5 \)) the friction loss is twice as large for rods with a roughness of 100 \( \mu \)m as for perfectly smooth rods [1]. The friction loss, however, accounts for only 1/6 of the total pressure drop. Therefore, a fourfold increase cannot be explained.

c) Due to nucleate boiling, a two phase flow condition had appeared in some fuel assemblies. According to Tong and Weisman [1], a flow quality of 5% causes a fourfold increase of the pressure drop. The flow quality is the share of the vapour flow in the total flow. At reactor conditions, a flow quality of 5% implies a void fraction of 20% [1]. This would be most uncommon for a pressurized water reactor!

Explanation c) is the most viable one, nevertheless. The effects on reactivity and on neutron noise also point to a substantial void fraction. This is discussed in sections 4, 5 and 6.

4. LOCAL REACTIVITY EFFECTS

A temperature increase and the appearance of vapour bubbles both have a negative effect on reactivity. At the end of life conditions, the reactivity parameters are:

\[
\begin{align*}
\text{Temperature coefficient} & : -70 \text{ pcm/°C} \\
\text{Void coefficient} & : -190 \text{ pcm/%}
\end{align*}
\]

A temperature increase of 10 °C and a void fraction of 20% therefore cause a reactivity effect:

\[
\Delta \rho = (-70 \text{ pcm/°C}) \times 10 \text{ °C} + (-190 \text{ pcm/%}) \times 20 \%
\]

\[
= -4500 \text{ pcm}
\]

Compared with -4500 pcm, the other reactivity effects such as Doppler and Xenon poisoning are negligible.

When the reactivity decreases in a fuel assembly, the neutron flux also decreases. One Group Theory yields the following simplified formula:

\[
\Delta \Phi = \frac{h^2}{4 \cdot M^2} \cdot \Delta \rho
\]

with

\[
\begin{align*}
h & : \text{Width of the fuel assembly (20 cm)} \\
M & : \text{Migration length of the neutrons (7 cm)} \\
\phi & : \text{Neutron flux} \\
\Delta \phi & : \text{Neutron flux difference between top and bottom}
\end{align*}
\]
hence

\[
\frac{A \Phi}{\Phi} = -9 \%
\]

This value agrees reasonably well with Fig. 3. At position G4 (oxidized cladding), the neutron flux in the upper half of the assembly is some 7% lower than at D5 (normal cladding).

5. GLOBAL REACTIVITY EFFECTS

a. Critical boron concentration

A reactivity loss of 4500 pcm in the upper half of 12 fuel assemblies causes a global reactivity loss of roughly

\[
\frac{12 \times 121}{12} \times \frac{1}{2} \times 4500 = 220 \text{ pcm}
\]

The critical boron concentration should therefore be some 25 ppm lower.

The concentration was actually lower than predicted, but this could also be due to the inaccuracy of the calculations. Core depletion tends to be underestimated at DOEL 1-2.

b. Axial offset

The reactivity loss in the upper half of the core causes a shift of the power distribution. The axial offset becomes more negative. Fig. 4 shows the calculated void fraction in the upper half of 12 fuel assemblies that accounts for the observed axial offset during cycle 11 (Fig. 1). The value of 19% on May 6, 1986 agrees with our crude thermohydraulic analysis.

c. Pressure coefficient of reactivity

The pressure coefficient of reactivity increases with the void fraction in the core. The reason is that vapor bubbles are more compressible than liquid. A void fraction of 20% in the upper half of 12 fuel assemblies causes a calculated increase of 1.2 pcm/bar.

On July 3, 1986 we measured the pressure coefficient of DOEL 2:

<table>
<thead>
<tr>
<th>Expected value</th>
<th>5.4 pcm/bar (Confirmed at DOEL 1)</th>
</tr>
</thead>
<tbody>
<tr>
<td>Value found</td>
<td>7.0 pcm/bar</td>
</tr>
<tr>
<td>Difference</td>
<td>1.6 pcm/bar</td>
</tr>
</tbody>
</table>

6. NEUTRON NOISE

On July 6, 1986 we positioned the mobile fission chambers at different locations in the core and we analysed the neutron noise. In the upper half of the suspected assemblies, the noise spectra showed a clear signature of nucleate boiling. Fig. 5 shows the spectra measured between the 1st and 2nd and between the 5th and 6th spacer grid of the assembly at position G4. Fig. 6 does the same for the assembly at position D7.

In the low frequency range, the normalised autocorrelation spectral density (NAPSD) shows the usual peaks at 4.5 and 9.2 Hz. They originate from the first two vibrational modes of the fuel assemblies.

Filtered white noise with a cut off frequency of 25 Hz seems superposed in the high frequency range (Up to 50 Hz).
Nucleate boiling

P. Bernard et al. observed a similar phenomenon during experiments on nucleate boiling in PWR fuel assemblies [2]. According to their theory, the amplitude of the "boiling" noise is proportional with the void fraction and the cut-off frequency is a function of the bubble speed in the vicinity of the detector. They derived the following formula for the "boiling" noise:

\[ \rho(f) = k \cdot \left( \frac{\pi \cdot \theta \cdot f}{\pi \cdot \theta \cdot f} \right)^2 \]

with

\( f \) : Frequency
\( k \) : Factor proportional with the void fraction
\( \theta \) : Transit time of the vapour bubbles in the vicinity of the detector

The theory was confirmed during experiments on the IRENE test loop set in the OSIRIS reactor at SACLAY (France).

Extrapolation to DOEL 2 yields:

- Void fraction : 10 ... 20 %
- Transit time : some 15 ms
- Average bubble speed : some 2 m/s

Those values are compatible with thermohydraulics and neutronics.

7. NUCLEATE BOILING

We attribute the increased nucleate boiling rate to the oxidized cladding of the third cycle fuel assemblies. The link between surface roughness and nucleate boiling is well established. The use of boiling stones in every laboratory is based hereupon.

P. Bernard et al. state in their paper [2]:

"The general purpose of the experiment was to collect informations and measurements about the importance of crud deposits resulting from chemical conditions and the nucleate boiling rate level. This program was defined and carried out in collaboration by CEA, EDF, FRAMATOME and WESTINGHOUSE".

Crud deposits on the fuel rods seem to promote nucleate boiling. Nucleate boiling on the other hand, seems to promote crud deposition, just as high burnup does [3]. This makes the process self-sustaining (Fig. 7).

Some aspects of the DOEL 2 phenomena remain puzzling, however:

a) Why at DOEL 2 and only at DOEL 2? Similar fuel was sold to Tihange 1 and several other foreign nuclear power plants. Up to this moment, no other core has behaved like DOEL 2.

b) Why did the phenomenon almost disappear after a cold shutdown and why did it reappear (Fig. 2)? Does a temperature transient affect the oxide layer on the fuel rods?

8. CYCLE 12 OF DOEL 2

Cycle 12 of DOEL 2 started on October 6, 1986. Some third cycle fuel assemblies with oxidized cladding were loaded. In order to reduce the silicon concentration in the primary circuit below 100 ppb, no recycled boron was used. We hoped that this would keep the phenomena from reappearing.

At a core burnup of 5500 MWD/Ton however, some thermocouple readings again went up.
9. CONCLUSIONS

a) The status of the reactor core must be monitored frequently by all available means. The safety margins might be reduced by hitherto unknown phenomena.

b) Neutron noise analysis can be used successfully to detect nucleate boiling in PWR fuel assemblies.

10. REFERENCES


11. ACKNOWLEDGEMENT

We are indebted to A. TIMMERMANS plant manager of DOEL 1-2, who granted permission to publish and to L. DE MEYER who prepared the drawings.
Nucleate boiling

Doel 2 Cycle 11
Axial offset as a function of burnup

Fig. 1

Doel 2 Cycle 11
Core exit temp. as a function of burnup

Fig. 2

Doel 2 Cycle 11
Flux map of May 6, 1986

Fig. 3

Doel 2 Cycle 11
Void fraction as a function of burnup

Fig. 4
ABSTRACT

In-core neutron detectors were used for noise diagnostic measurement in a commercial PWR NPP. Phase dependence versus frequency between axially placed self powered neutron detectors (SPND) exhibited linear behaviour in some cases. Correlation was found between the existence of linear phase behaviour and the power of the Unit. More thoughtful consideration showed a dependence on local power, and on overheating of the surface of fuel elements. Data extracted from conventional measurement of top of core temperature sensors and in-core SPNDs, using the conventional system for linear power production estimation showed that linear phase takes place where linear power production is over 2 MW/m. One dimensional thermohydraulic calculation pointed out that at those places the surface temperature is near or over the saturation value. Thus the appearance of linear phase versus frequency can be attributed to phenomena of local subcooled boiling and therefore its existence can be used for boiling detection in PWRs. All basic considerations had been checked beforehand in model experiments and are cited here.

1. INTRODUCTION

The effect of void transport on neutron fluctuation measured by in-core neutron detectors was first observed in boiling water reactors (BWRs) (Wach, 1973; Kosaly, 1977; Seifritz, 1973), showing a near linear dependence of phase of the crosspower spectral density function (CPFD) on frequency between axially placed neutron detectors. This phenomenon was described phenomenologically by Wach and Kosaly (Wach, 1974) who introduced the so-called global and local conception for this noise effect. This simple theory well explained almost all details of the measurement including the undulation of phase dependence on frequency around the linearity. Most of the theories introduced since that simply placed that phenomenological model into a more sophisticated model, such as into a coupled point kinetic neutron and one-dimensional thermohydraulic model (Katona, 1982), but they did not advance the explanation of the effect too far. One of the most thoroughly examined questions was and still remains the detector field of view for the local effect. Many authors have contributed to this question both theoretically and experimentally. The introduction the concept of far and near field noise (Kosaly, 1982) made the old picture more clear, and the exponential detector
field of view introduced by the Delft school has made a considerable progress in that field (Kleiss, 1979).

The effect of propagating void on neutron field can be regarded as well examined today, nevertheless it can be applied only to estimate the propagating velocity, but the other important question, viz. the determination of void fraction from these measurements, remains unsolved at least on industrial scale, despite numerous theoretical and experimental efforts.

The effect of propagating noise sources on neutron field in pressurized water reactors (PWRs) has remained untouched for somewhat, because no linear phase of CPSD was observed between axially placed neutron detectors. Perhaps one of the first well published direct pieces of evidence for a local effect in PWR but still smaller than the global one proving the acceptance of Wach-Kosaly's formula, was the phase behaviour found by Por and Turkcan in Borssele PWR (Por, 1981; Katona, 1982). In a long run experiment, which had therefore very small statistical error, a clear undulation of the phase of CPSD between axially placed in-core neutron detectors was found, and it was also demonstrated that the amplitude of undulation depends on boron concentration, i.e. on moderating-absorbing capability of the propagating coolant (see Fig. 1). An explanation was derived in the frame of one dimensional coupled neutron-thermohydraulic model (Katona, 1982). Thus there was a clear evidence that propagating disturbances affect neutron noise in PWRs as well, but in the absence of the void no linear phase behaviour was found.

![Graphs showing phase shift vs. frequency for boron concentrations of 1738 ppm and 748 ppm.]

**Fig. 1.** Dependence of phase on frequency between axially placed in-core neutron detectors in Borssele PWR at different boron concentration (from Por, 1981)

The Rheinsberg experiment (Rindelhardt, 1985) was the one which was most directly aimed at proving experimentally that these two effects are the same and the appearance of the linear phase can be connected with the appearance of boiling in PWRs. In specially designed experimental assemblies (EK) (Krause, 1986) containing an additional remote controlled valve at their inlet, the coolant flow could be decreased through EK and thus a real boiling was achieved in a commercial working 70MWe PWR. The thermohydraulic conditions were well controlled by numerous built-in thermocouples and differential pressure transducers. Thus determination of the onset and the place of boiling could be carried out in an independent way. As it was apparent from published results (Fig. 2.) (Liewers, 1985), the boiling phenomenon and the linear phase behaviour versus frequency were linked with each other. Furthermore, a connection between the void fraction and the measured low frequency neutron noise signals could be identified and described in a combined neutron and thermohydraulic model (Collatz, 1986).
Fig. 2. Changes of phase and coherence between axially placed SPNDs with flow rate decreasing up to boiling point in Rheinsberg experiment (Rindelhardt, 1985).

Fig. 3. Coherence appeared in model experiment only at onset of boiling (Aguilar, 1987; Por, 1985).

Another item of experimental evidence was published by Aguilar and Por (Aguilar, 1987; Por, 1985). In an experiment carried out in a critical assembly an independent heat source was used to boil water circulating in a test loop through the core. Coherence and signal transmission path analysis (Fig. 3.) have proved that temperature noise affects neutron noise solely through boiling bubbles, while in the absence of void checked by optically there was no connection between temperature and neutron fluctuation.

2. NOISE MEASUREMENT IN PAKS PWR

All Units of Paks Nuclear Power Station have been equipped with noise measuring and evaluating systems (see details and concepts of these gradually enlarged and developed systems in Hollo, 1982; Valkó, 1985; Por, 1986; Valkó, 1987). Unit 2 has noise measuring lines in primary loops as follows: 41 vibration transducers positioned at different main components, 11 pressure transducers, two top-of-core thermocouples, as many as 6+2 indirectly and directly coupled ex-core ionization chambers and, most importantly, there are three strings of axially placed in-core self powered neutron detectors (SPND) (see Fig. 6a). Each string contains seven SPNDs one above the other with emitters of 21 cm made from Rh, and one common compensation cable for g-background correction. For Unit 1 we had only one string of SPNDs. At the beginning of its operation the phase of CPSD between axially placed SPNDs measured by that string showed the so-called undulation around the zero with growing frequency between 0 and 4 Hz (see Fig. 4). After refueling Unit 1, when its nominal power had been increased to 110% of nominal due to thermo-hydraulic means, linear phase versus frequency was found from time to time between the same SPNDs (Fig. 5.).
Fig. 4. Typical coherence and phase measured between axially placed SPNDs in Paks NPP Unit 1 during the first fuel cycle.

Fig. 5. Phase and coherence between axially placed SPNDs in Paks NPP Unit 1 during after refueling.

Measurements were carried out regularly in each month at both units. At Unit 2 where we had three strings, linear phases versus frequency were found from the very beginning of its operation. But these linear phases do not appear in each string and even not between each pair of detectors. A typical example of this situation is reproduced as a collection of phases and coherences in Fig. 6b. The three strings have the following core coordinates: 04-37; 11-54; and 15-32. This record was made on 27th of May, 1986. Table 1 contains the plant data of that day.
Table 1  Most important data of core of Unit 2 on 27 May 1986

<table>
<thead>
<tr>
<th>Parameter</th>
<th>Value</th>
</tr>
</thead>
<tbody>
<tr>
<td>Thermal Power</td>
<td>1364.1 MW</td>
</tr>
<tr>
<td>Flow rate through core</td>
<td>29005.2 t/h</td>
</tr>
<tr>
<td>Pressure in the core</td>
<td>122.4 bar</td>
</tr>
<tr>
<td>Pressure drop on the core</td>
<td>2.4 bar</td>
</tr>
<tr>
<td>Inlet temperature</td>
<td>264.5 °C</td>
</tr>
<tr>
<td>Average temperature rise in the core</td>
<td>32.1 °C</td>
</tr>
<tr>
<td>Average linear energy production</td>
<td>1.6 MW/m</td>
</tr>
<tr>
<td>Boron concentration</td>
<td>1.5 g/kg</td>
</tr>
<tr>
<td>Position of control rods</td>
<td>187.5 cm</td>
</tr>
</tbody>
</table>

Fig.6a. Arrangement of in-core and excore neutron detectors and thermocouples in Unit 2 of Paks NPP noise measuring system (with indication of elevations in our calculations)
Fig. 6b. Coherence and phase between different pairs of SPNDs of different coolant channels (assemblies) in Paks NPP Unit 2 at 27 of May, 1986.
While there is only slight undulation of phase around zero between detectors of string 04-37, it has become almost linear in string 11-54 and certainly it has a good linearity for string 15-32 - at least for detectors in the middle. There are two more experimental observations in phase pictures: No linearity was found for the top two detectors in string 15-32; and an inverse start of phase (at low frequencies) versus frequency was found for some combinations with the lowest SPND.

In the light of our previous experience discussed in the Introduction, it was proposed that onset of boiling was achieved when the power of Unit 1 had been increased and also in Unit 2. But the average plant data do not allow one to make a conclusion on the existence of a real boiling process in the core and also the variety of appearance was disturbing in terms of that explanation. Therefore the first suggestion was that a mechanism other than boiling should be considered to be responsible for linear phase behaviour.

3. MEASURING AND CALCULATING POWER PRODUCTION

Units are equipped with sophisticated in-core monitoring systems called VERONA (WWER Reactor On-line Analysing System)(Por,1986). The system is based on the available standard in-core instrumentation and using a TPA 1148 type computer, it is capable of monitoring all fuel assembly thermocouples, all neutron signals, position meters of the control rods, and about 100 signals from the technology. The system evaluates many different parameters and distribution profiles. One of these is the linear energy production calculated for ten elevations axially in each fuel assembly, based mainly on dc values measured by SPNDs but also using flux distribution, thermocouple measurements and actual plant data. Table 2 contains the actual calculated linear power production for 27 May 1986 during the noise measurement for our three fuel assemblies.

Table 2. Measured linear power production for different assemblies at 27 May 1986

<table>
<thead>
<tr>
<th>Number of assembly</th>
<th>04-37 (MW/m)</th>
<th>11-54 (MW/m)</th>
<th>15-32 (MW/m)</th>
</tr>
</thead>
<tbody>
<tr>
<td>Elevation</td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>1</td>
<td>0.98</td>
<td>1.22</td>
<td>1.37</td>
</tr>
<tr>
<td>2</td>
<td>1.49</td>
<td>1.86</td>
<td>2.09</td>
</tr>
<tr>
<td>3</td>
<td>1.58</td>
<td>1.98</td>
<td>2.22</td>
</tr>
<tr>
<td>4</td>
<td>1.60</td>
<td>2.02</td>
<td>2.26</td>
</tr>
<tr>
<td>5</td>
<td>1.61</td>
<td>2.06</td>
<td>2.28</td>
</tr>
<tr>
<td>6</td>
<td>1.63</td>
<td>2.09</td>
<td>2.30</td>
</tr>
<tr>
<td>7</td>
<td>1.59</td>
<td>1.99</td>
<td>2.27</td>
</tr>
<tr>
<td>8</td>
<td>1.45</td>
<td>1.77</td>
<td>2.15</td>
</tr>
<tr>
<td>9</td>
<td>1.20</td>
<td>1.44</td>
<td>1.85</td>
</tr>
<tr>
<td>10</td>
<td>0.69</td>
<td>0.82</td>
<td>1.09</td>
</tr>
<tr>
<td>Average</td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>Temperature rise °C</td>
<td>27.6</td>
<td>36.8</td>
<td>39.7</td>
</tr>
<tr>
<td>Inequality (k)</td>
<td>0.85</td>
<td>1.16</td>
<td>1.26</td>
</tr>
<tr>
<td>Power produced in ass.(MW)</td>
<td>3.47</td>
<td>4.48</td>
<td>4.98</td>
</tr>
</tbody>
</table>

Based on data measured by the VERONA system, thermohydraulic calculations were carried out to check the real thermohydraulic conditions during the measurements. The program(Rhode,1986) which was used for that was a one channel thermohydraulic code which starts from the measured data of inlet temperature, total power of the given fuel assembly and the axial distribution of the power. Also the actual coolant flow rate and the primary system pressure were taken into account. To consider the radial micro-power distribution in the fuel assembly, realistic load factors (k=1.0;1.2;1.3) were used for the calculations.
Table 3. Results of thermohydraulic calculations for assembly 15-32

<table>
<thead>
<tr>
<th>Elevation</th>
<th>$T_{clad}$ (°C)</th>
<th>$T_{cool}$ (°C)</th>
<th>$x$</th>
<th>$T_{clad}$ (°C)</th>
<th>$T_{cool}$ (°C)</th>
<th>$x$</th>
</tr>
</thead>
<tbody>
<tr>
<td>1</td>
<td>279</td>
<td>265</td>
<td>0</td>
<td>283</td>
<td>266</td>
<td>0</td>
</tr>
<tr>
<td>2</td>
<td>289</td>
<td>269</td>
<td>0</td>
<td>297</td>
<td>271</td>
<td>0</td>
</tr>
<tr>
<td>3</td>
<td>295</td>
<td>274</td>
<td>0</td>
<td>304</td>
<td>277</td>
<td>0</td>
</tr>
<tr>
<td>4</td>
<td>300</td>
<td>279</td>
<td>0</td>
<td>311</td>
<td>283</td>
<td>0</td>
</tr>
<tr>
<td>5</td>
<td>305</td>
<td>283</td>
<td>0</td>
<td>317</td>
<td>289</td>
<td>0</td>
</tr>
<tr>
<td>6</td>
<td>310</td>
<td>288</td>
<td>0</td>
<td>323</td>
<td>295</td>
<td>0</td>
</tr>
<tr>
<td>7</td>
<td>314</td>
<td>293</td>
<td>0</td>
<td>328</td>
<td>301</td>
<td>0</td>
</tr>
<tr>
<td>8</td>
<td>317</td>
<td>297</td>
<td>0</td>
<td>331</td>
<td>306</td>
<td>0</td>
</tr>
<tr>
<td>9</td>
<td>318</td>
<td>301</td>
<td>0</td>
<td>331</td>
<td>311</td>
<td>0.5</td>
</tr>
<tr>
<td>10</td>
<td>314</td>
<td>304</td>
<td>0</td>
<td>327</td>
<td>314</td>
<td>0.9</td>
</tr>
</tbody>
</table>

where: $k$ - coeff. inequality inside fuel assembly
$x$ - void containt in the coolant in percent

In Table 3 the main results of the calculations (the cladding temperature, coolant temperature and void fraction) are given for the load factors of $k=1.0$ and $1.3$. The saturation temperature was $326°C$.

4. DISCUSSION

If one compares the results of VERONA estimation of linear energy production (Table 2) with measured phase behaviours (Fig.6) it is apparent that the linear phase was measured where linear energy production was around above 2 MW/m. It is worth mentioning that linearity of the phase shows correlation with the total energy production of given fuel assembly. Because the radial power production in the core changes during the core life time, the appearance and disappearance of linear phase can be explained by the changing load factors of the three used fuel assemblies.

The result of the thermohydraulic calculation show (Table 3) that if $k$ is assumed to be as much as 1.3 then real voids can appear in the most heavily loaded fuel assembly at a certain spot. It is interesting that even in this case only the temperature of the cladding was above the saturation temperature ($T_{sat}=326°C$) beginning from the 7th elevation. But even that falls towards the 10th elevation, i.e. at the top of the fuel assembly the cladding temperature is below of the boiling temperatue of the coolant. The mean coolant temperature even in the most heavily loaded assembly at the flux maximum was lower than the saturation temperature. This means that no real (volume) boiling exists in the fuel assembly. But it is well known from the theory of the mechanism of void formation, that if there is an overheated surface, bubbles will appear on it well below the saturation temperature either due to its roughness or due to other local irregularities which can cause overheating.

These steam bubbles can leave the surface and will subsequently collapse in the subcooled coolant. Their lifetime follows random rule according to their randomly changing circumstances. But during this time they travel some distance with the coolant. Even if they collapses immediately they will cause strong temperature fluctuations in the subcooled fluid which will travel over long distances. This is what affects the neutron field, this is the local transport effect, which is measurable by in-core neutron detectors. Now it is understandable why we did not measure linear phase at the top of string 15-32. Bubbles could not reach the top.
of the fuel assembly, they collapsed earlier, in spite of the fact that the temperature of the coolant there was higher. But since no overheated surface of cladding existed there, bubble formation was excluded.

From this it is also obvious that in those PWRs where real or subcooled boiling is not supposed to take place the registration of linear phase behaviour between axially placed neutron detectors at different places of the core can be used for detecting so called "hot spots" of the fuel. This is very important for monitoring fuel elements, evaluating their possible core lifetime, i.e. for estimation of optimum reloading of the core. For that reason one needs SPNDs as much as possible scattered all over the core. WWER type PWRs have the advantage of 36 strings of SPNDs (7 detectors in each, i.e. in total 252 SPNSs) equally distributed in the core. In future units of WWER-NPP, there will be the opportunity to reach all those SPNDs by noise diagnostic systems.

6. ACKNOWLEDGEMENTS

Measurements were carried out at Paks NPP with wide cooperation of the noise diagnostic team. Their assistance and the kind permission to publish these result are highly appreciated. The thermohydraulic calculations was performed by U. Rhode whose contribution is greatly acknowledged.

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INTERPRETATION OF VELOCITIES DETERMINED BY NOISE ANALYSIS FOR VARIOUS VOID FRACTIONS AND FLOW REGIMES IN TWO-PHASE FLOW

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Abstract - Experiments are performed in an air-water loop in order to gain insight into the physical interpretation of the velocities measured by noise correlation in a BWR. Thermal neutrons and detectors are modelled by visible light produced in the water and phototransistors. Bubbly, slug and froth flow are examined. The flow type at issue in the loop can be determined from the probability density function of the photo-transistor signals.

For each flow pattern the measured velocity is much higher than the volume averaged water and air velocity and the volumetric flux, which are usually taken as an interpretation. However, for bubbly flow it corresponds with the calculated velocity of bubbles without wall interaction. In the case of slug flow it is indicated by an independent experimental method that it is the average slug velocity that is measured by noise correlation. For froth flow no theoretical basis is found.

1. INTRODUCTION

Cross correlation of the noise of the signals of axially separated incore detectors is a widely used technique to obtain information on the velocity of the two-phase mixture through the core of a BWR. The physical interpretation of the measured velocity, however, has been an important topic of discussion for some time: it is not clear whether the steam velocity or the water velocity or something in between (for instance the volumetric flux (Lübbesmeyer, 1983)) is measured. The situation is strongly complicated by the existence of various flow regimes - as the void fraction varies from 0% at the bottom to 70% at the top of the core, regimes from steam bubbles in water to droplets in steam are to be expected. Nevertheless it is of great importance to be able to measure the coolant velocity with satisfying accuracy for several applications:

- the coolant flow in a natural circulation cooled BWR is not known and can only be derived from thermalhydraulic calculations or design values
- the distribution of the coolant over the core cannot be measured otherwise without disturbing the coolant flow
- a possibility would be offered to verify computer codes and input data for these codes.

Another issue is the possibility of determining the flow regime under consideration from the noise signals or from their characteristic functions (probability density function, correlation function, power spectrum, coherence) (Albrecht et al., 1982).
A clear review on both topics is given by Lübbesmeyer (1984).

The present paper deals with experiments on an air-water loop that models a coolant channel of a BWR. Unlike earlier simulations the set-up models the moderation effect of neutrons as well. In future the set-up can be extended to study subjects as the influence of subchannel-velocities, void drift etc.

2. DESCRIPTION OF THE AIR-WATER LOOP

A coolant channel of a BWR is modelled by a glass tube two meters in height through which water is pumped upwards. The inner diameter of the tube is 3.66 cm. Air can be added at the bottom of the tube to obtain a void fraction as high as 70%. The air and water flow can be
changed separately thus making it possible to produce bubbly, slug or froth flow. The latter flow pattern is also called churn, wave entrainment, dispersed slug or semiannular. A detailed description of these flow types can be found in textbooks like Govier and Aziz (1972). UV-light sources (wavelength = 370 nm, 1 meter in height) are placed at two opposite sides of the tube; their radiation excites a fluorescent powder that is dissolved in the water and that emits visible light (wavelength = 560 nm) when shone upon. Photo-transistors that are fixed in rings around the tube register the emitted visible light and serve as a model for neutron or gamma detectors, depending on their field of view (Van Dam, 1976 and Van Dam and Kleiss, 1985). They are not sensitive to the UV-light analogous to neutron detectors that do not detect fast neutrons. In this manner the signal carrier - here visible light - is produced in the water as thermal neutrons are in the coolant of a reactor. The UV-radiation models the fast neutrons in a reactor. Figure 1 gives an impression of the set-up.

![Diagram](image)

**Fig. 1.** The experimental set-up.

The relaxation lengths \((\lambda)\) of the two types of light were adjusted by varying the concentration of the fluorescent powder and by adding a powder that absorbs the visible radiation. For the measurements given below values of 5.5 cm for the UV light and 2.5 - 3.0 cm for the visible light were chosen. The latter value corresponds with the relaxation length \(\ell\) that characterizes the local field of view of a neutron detector in a BWR, being (Van Dam, 1976)

\[
\lambda = \frac{1}{\frac{1}{\ell^2} + \frac{1}{\tau}}
\]

with \(\ell = \text{thermal diffusion length}\) and \(\tau = \text{Fermi age}\). The relaxation lengths were checked by measuring the attenuation of the two light types with a photo-multiplier (selecting the wavelength with a monochromator). The results correspond nicely with the desired values

\[
\begin{align*}
\lambda_{\text{UV}} &= 5.5 \pm 0.2 \text{ cm} \\
\lambda_{\text{visible}} &= 2.7 \pm 0.1 \text{ cm}
\end{align*}
\]
The physical properties of the water are not changed significantly by adding the powders to the water: the specific density was found to be 1000 kg/m³ and the surface tension 70 \((\pm)10^{-1}\) N/m (water (23 °C) 72x10⁻² N/m).

3. CHARACTERISTIC FUNCTIONS

In order to try to identify the flow pattern at issue the following characteristic functions from the noise signals might be considered:

- the probability density function (PDF) (Albrecht et al., 1982)
- the cross-correlation function (CCF)
- the auto and cross power spectral density (APSD, CPSD)
- the coherence spectrum \( \gamma^2 \).

For the definition and a mathematical outline of these functions see Jenkins and Watts (1968).

A similar attempt - covering a great number of flow regimes - was performed with limited success by Lübbesmeyer (1984).

Fig. 2 displays the characteristic functions for the three observed flow regimes: bubbly, slug and froth.

The PDFs show that bubbly flow is characterized by one peak (unimodal flow) that is due to small changes of the signal around the mean value by passing bubbles. As air bubbles do not contain the fluorescent powder they do not produce visible light and the signal of the photo-transistor will decrease. In the case of slug flow the PDF consists of one distinct peak at a negative signal value caused by air slugs that nearly fill the entire cross-section of the tube and a broad maximum at a positive value formed by small bubbles in water. The PDF for froth flow exhibits two distinct peaks; one peak is due to the passing of a big air bubble and the other to the passing of a liquid slug (bimodal flow). The usage of this unimodal/bimodal character of the PDF as an identification method in a BWR is impeded in two ways:

- the low-frequent behaviour is disturbed by the global neutron noise. This problem can be overcome by subtracting the signal of an excorine neutron detector that only measures the global noise component (Albrecht et al., 1982).
- the signal of a neutron detector is influenced by the two-phase flow through the rod-bundles surrounding the detector. The situation in case of a gamma detector is even more complicated as its field of view is larger (Van Dam and Kleiss, 1985).

The APSD in case of bubbly flow is flat and the coherence low. Due to this low coherence the linearity of the phase spectrum - indicating a transit time, see section 5 - is lost for frequencies higher than 10 Hz. For other flows the spectra decrease strongly with increasing frequency and the coherence is much higher. The difference in phase angle for the three flow types denotes different velocities (Section 5). Bubbly flow yields the lowest measured velocity, froth flow the highest.

The CCF exhibits for all flow patterns a distinct maximum that can be used to determine the velocity (Section 5). In the case of bubbly flow the CCF-maximum is low due to the low coherence.

In conclusion it can be stated that the flow regime in this test loop can be distinguished by observing the PDF. In order to allow for meaningful application in a BWR, however, more research, using more complex set-ups, is required. Therefore the model will be extended in the future.

4. THEORETICAL VELOCITIES

In order to gain insight into the measured velocities the air and the water velocity must be known. The average velocities \( V_a \) and \( V_w \) can be derived from the void fraction \( \alpha \) and the volume flow rates of the air \( Q_a \) and the water \( Q_w \), according to

\[
V_a = \frac{Q_a}{\alpha A} \quad \text{(3)}
\]

\[
V_w = \frac{Q_w}{(1-\alpha)A} \quad \text{(4)}
\]

in which \( A \) is the cross-section of the tube.
Fig. 2. Flow patterns and characteristic functions for bubbly, slug and froth flow.
The void fraction was determined by measuring the level of the water in the reservoir by means of a gauge (Fig. 1) and comparing this with the level without air. As the total volume of the tube above the air inlet is known, the average void fraction can easily be derived. In case of bubbly flow, the void fraction obtained in this way was checked with a gamma-absorption technique. A 300 mCi $^{113}$Am gamma source and a NaI scintillation-detector were placed at opposite sides of the tube. The void fraction can be calculated from the measured gamma intensity when the intensities in the situation of a tube filled with water and a tube filled with air are known. This measurement was performed at three axial positions. The results are reported in Table 1; the void fraction obtained by the level measurement for this case is $24 \pm 5\%$.

<table>
<thead>
<tr>
<th>distance from the</th>
<th>$a$ (%)</th>
<th>$\sigma_a$</th>
</tr>
</thead>
<tbody>
<tr>
<td>air inlet (cm)</td>
<td></td>
<td></td>
</tr>
<tr>
<td>36</td>
<td>23.72</td>
<td>0.05</td>
</tr>
<tr>
<td>63.5</td>
<td>22.51</td>
<td>0.10</td>
</tr>
<tr>
<td>91</td>
<td>23.04</td>
<td>0.53</td>
</tr>
</tbody>
</table>

From this table it can be concluded that the void fraction does not vary significantly along the axis of the tube and that it corresponds with the fraction obtained by the measurement using the water level. This validates the latter method.

Note: it is quite complicated to measure the void fraction for the other flow types using the gamma-absorption technique since in those cases the void is not uniformly distributed over the cross-section of the tube. Therefore the level method was used throughout.

The flow rates were measured by means of Rota-flowmeters. In order to calculate the precise airflow rate a pressure correction has to be carried out. Therefore a manometer was installed right behind the flowmeter. The measured volume flow rate has to be corrected using

$$Q_a = Q_{a,m} \cdot \left( \frac{P_{cal}}{P_m} \right)^{1/2}$$

(5)

In which $Q_a$ = the corrected volume flow rate through the flowmeter

$Q_{a,m}$ = the volume flow rate indicated by the flowmeter

$P_{cal}$ = the pressure at which the flowmeter was calibrated = $1.18 \times 10^5$ N/m²

$P_m$ = the pressure of the air in the flowmeter

(measured by means of the manometer).

$Q$ does not represent the volume flow rate of the air through the tube as the pressure drops significantly in the air inlet.

As for the pressure of the air at the position of the photo-transistors a second correction has to be performed. For this purpose the pressure gradient of the air-water mixture was calculated, using (Govier and Aziz, 1972, page 348)

$$\frac{\Delta p}{\Delta z} = \frac{\rho_a V_w (1-a)}{V_M} \left( \frac{\rho_w}{\rho_a} \right) \frac{g}{D} f_{tp}$$

(6)

in which $\Delta p/\Delta z$ = pressure drop per unit length

$\rho_a$ = density of the air

$\rho_w$ = density of the water

$V_M$ = volumetric flux = $a V_a + (1-a) V_w = (Q_a + Q_w)/A$

(average velocity of the mixture)

$g$ = acceleration due to gravity

$D$ = tube diameter

$$\frac{\Delta p}{\Delta z} = \frac{\rho_a V_w (1-a)}{V_M} \left( \frac{\rho_w}{\rho_a} \right) \frac{g}{D} f_{tp}$$

(6)
\[ f_{tp} = \text{two-phase friction factor.} \]

This formula holds for all flow types except annular mist. Experimental data on the two-phase friction factor were obtained from Gouvier and Aziz (1972). The pressure at the position of the photo-transistors was calculated using \( p_{out} = 1 \text{ atm} (= 1.01 \times 10^5 \text{ N/m}^2) \) and outlet velocities for \( V_a \) and \( V_w \). The volume flow rate of the air at this position \( (Q_{a,p}) \) is obtained with

\[
Q_{a,p} = Q_a \cdot \frac{P_m}{P_{out}} \cdot \frac{P_{out}}{P_{out} + \frac{AP}{Az} \cdot z_p} = Q_{a,m} \cdot \frac{\sqrt{P_{cal}P_m}}{P_{out} + \frac{AP}{Az} \cdot z_p}
\]  

(8)

In which \( z_p \) denotes the distance from the photo-transistors to the outlet of the tube. With the \( P_{cal} \) of Eqs. (3-8) the average air velocity \( V_a \) and thus \( V_M \) at the position of the photo-transistors can be calculated.

The velocity profile of the bubbles over the tube cross-section is not flat; bubbles near the tube wall will have a considerably lower velocity than those in the centre, due to wall interaction. Wall interaction can be neglected for bubbles in the centre of the tube (Wallis, 1969, page 251). Their velocity can be calculated using the drift flux model (neglecting wall interaction):

\[
V_b = C_o V_M + V_{bs}
\]  

(9)

with \( V_b \) = the bubble velocity

\[
C_o = \frac{<V_M>}{<\omega> <V_M>}, \text{ where } < \cdot > \text{ denotes averages over the tube cross section}
\]

\( V_{bs} \) = the bubble velocity in stagnant water.

For bubbly flow \( 1.0 < C_o < 1.5 \) holds, with a most probable value of 1.2 (Wallis, 1969, page 256). The bubble velocity \( V_b \) has to be corrected for the void fraction:

\[
V_{bs} = f_a V_{bso}
\]  

(10)

In which \( V_{bso} \) = the velocity of a single bubble in stagnant water

\[ f_a = \text{a factor that expresses the fact that the velocity of a swarm of bubbles is significantly less than that of a single bubble. Experimental data for several void fractions were taken from Gouvier and Aziz (1972).} \]

The velocity of a single bubble in stagnant water is expressed by (Gouvier and Aziz, 1972, page 367 or Wallis, 1969, page 250)

\[
V_{bso} = 0.33 g \cdot 0.76 \left( \frac{\mu_1}{\mu_M} \right)^{0.52} \left( \frac{d_b}{2} \right)^{1.28}
\]  

(11)

In which \( \mu \) denotes the dynamic viscosity of water [Ns/m²] and \( d_b \) the bubble diameter [m]. Note: all values have to be expressed in SI-units.

The diameter of bubbles close to the wall at the position of the photo-transistors was measured using a stroboscope and found to be 1.6 - 2.3 mm.

The velocity of the air-slugs \( V_s \) in case of slug flow is much higher than the volume averaged air velocity \( V_{a,m} \) as the small bubbles in between the slugs have a much lower (or even negative) velocity. The derivation of \( V_s \) Eq. (9) has to be adapted:

\[
V_s = C_o V_M + C_s V_{ss}
\]  

(12)

In which \( V_{ss} \) denotes the slug velocity in stagnant water. \( C_s \) describes the fact that the velocity of a slug is influenced by the wake of the preceding slug (Wallis, 1969, page 292). For \( V_{ss} \) the relation

\[
V_{ss} = 0.346 \sqrt{\mu_1}
\]  

(13)

can be used (Gouvier and Aziz, 1972, page 396). For a circular pipe and fully developed flow holds
\[ C_0 = 1.2 \text{ and } C_2 = 1 \quad (Re_M = \frac{V_M D_p}{\mu_w} > 8000) \]  

(14)

The resulting slug velocity is given by

\[ V_s = 1.2 V_M + 0.207 \quad \text{m/s} \]  

(15)

Due to its extremely turbulent nature froth flow is not described theoretically; no relevant experimental studies of this flow pattern are known to the authors. At present it will be regarded as a modified form of slug flow.

Table 2 displays the theoretical velocities that are considered for comparison with the velocities measured by noise correlation. The void fraction is assumed to be constant throughout the length of the tube. Thus only the air velocity is affected by the difference in pressure at the tube inlet and outlet. The bubble velocity \( V_b \) is calculated for a bubble diameter of 1.6 and 2.3 mm, using Eqs. (9-11), assuming the bubble diameter to be constant over the tube cross-section. The 'slug' velocities for froth flow are given in brackets as the used Equation (15) is not proved to be valid for this flow pattern.

Table 2. Theoretical and experimental velocities (cm/s).

<table>
<thead>
<tr>
<th>Flow Type</th>
<th>( \alpha (%) )</th>
<th>Theoretical Velocities</th>
<th>Experimental Velocities</th>
</tr>
</thead>
<tbody>
<tr>
<td></td>
<td></td>
<td>( V_w )</td>
<td>( V_b )</td>
</tr>
<tr>
<td>bubblly</td>
<td>33</td>
<td>20</td>
<td>27</td>
</tr>
<tr>
<td></td>
<td>34</td>
<td>15</td>
<td>26</td>
</tr>
<tr>
<td>slug</td>
<td>38</td>
<td>21</td>
<td>48</td>
</tr>
<tr>
<td></td>
<td>45</td>
<td>24</td>
<td>51</td>
</tr>
<tr>
<td>froth</td>
<td>55</td>
<td>29</td>
<td>112</td>
</tr>
<tr>
<td></td>
<td>77</td>
<td>57</td>
<td>116</td>
</tr>
</tbody>
</table>

5. VELOCITIES OBTAINED BY NOISE CORRELATION

To derive the velocity of the signal disturbance from the noise component of the signals of two axially separated detectors two methods are commonly used (Thie, 1981)

- determine the CCF of the two signals. This function has its maximum at the lag-time equal to the time that it takes the signal disturbance to go from the lower to the upper detector
- determine the phase spectrum of the complex CPSD of the two noise signals. The transition time is given by its linear slope.

As the distance between the two detectors is known the velocity of the signal disturbance can be derived.

Both methods will be used here. The maximum of the CCF is determined by fitting the maximum of the discrete CCF and its two neighbouring values to a parabola. A phase-extrapolation code is used to determine the phase for values higher than 180°. After that the phase spectrum is fitted to a straight line by means of a least squares method using the variance of the phase (that can be derived from the coherence) as a weighting factor.

The linearity of the phase is considerably influenced by the type of data- (or its equivalent lag-) window used, a fact which is often overlooked. For a detailed description of the use of windows see Harris (1978). Lag-windows symmetrical about \( t=0 \) are quite commonly in use. They were designed to obtain a better estimate of auto spectra. For the determination of the cross spectrum in case of a transit-time, however, such a window disturbs the symmetry of the CCF and benefits small delays. The estimated velocity determined from the CCF will be too high and the phase spectrum will be distorted only as a symmetrical CCF leads to a linear phase. It is clear that a lag-window has to be used that is symmetrical about the expected transit-time. The use of such a window is called alignment (Priestley, 1981). The influence of alignment to the phase and coherence spectrum is demonstrated in Fig. 3.

Table 2 presents the velocities derived from the fitted phase spectrum and the fitted CCF. A Hanning data-window in combination with alignment was used to determine the CCF and the CPSD. The difference between these velocities (denoted by \( AV \) in Table 2) is less than 1% for bubble and slug flow. However, using the phase spectrum might be disadvantageous as the
choice of the frequency-interval over which the phase is fitted to a straight line is subjective.

Fig. 3. Cross-correlation function (CCF), phase and coherence ($\gamma^2$) of two photo-transistor signals.

a) without alignment: $V_{CCF} = 77.2$ cm/s, $v_{phase} = 77.7$ cm/s
b) with alignment: $V_{CCF} = 74.6$ cm/s, $v_{phase} = 75.4$ cm/s.

Fig. 4. Theoretical water velocity ($V_w$), air velocity ($V_a$), volumetric flux ($V_m$), bubble velocity ($V_b$) and slug velocity ($V_s$) compared with the velocity derived by noise correlation ($V_{CCF}$).
A comparison between the theoretically derived velocities and the velocities obtained by fitting the CCF is given in Fig. 4. It is clear from this figure and from Table 2 that the measured velocity is in all cases considerably higher than $V_w$, $V_a$, and $V_m$ (which are brought out as interpretation by other authors). For bubbly flow the measured velocity is equal to the calculated velocity of bubbles without wall interaction ($V_b$). Apparently the centre of the tube, where there is no wall interaction, is the dominant part to the experimental velocity which is formed by a complicated mixture of the relaxation length of the light, the detector field of view (the field of view is very narrow near the tube wall, where the slow bubbles are rising), bubble diameter and coherence between the two signals. The velocity of bubbles near the tube wall was measured visually by means of a stop-watch. The measurement was quite accurate ($a_b < 3\%$) and resulted in $V_b = 28$ cm/s ($a = 33\%$) and $V_b = 27$ cm/s ($a = 34\%$). This indicates that indeed the velocity of those bubbles is very much influenced by the tube wall. The average air velocity $V_a$ is low due to the large contribution of the velocity of those bubbles. The unexpected fact that $V_a$ is actually lower than $V_b$ near the tube wall is attributed to measuring inaccuracies or the existence of smaller bubbles, whose velocity will be lower than the measured $V_b$.

The measured slug velocity $V_{slg}$ is higher than the theoretical velocity $V_m$. This is attributed to the fact that the slug pattern in the tube consists of fully developed as well as undeveloped slugs, which rise faster. The theory describes developed slugs, whereas $V_{slg}$ is an average of the velocities of developed and undeveloped slugs, and is therefore higher than $V_m$. The slug velocities were measured visually by making recordings with a video-camera and by using a stop-watch. This resulted in a range of velocities ($53-66$ cm/s for $a = 38\%$ and $61-79$ cm/s for $a = 45\%$) that covers $V_s$ and $V_{CCF}$.

6. CONCLUSIONS

The velocities measured by noise correlation in an air-water loop are much higher than the volume averaged water and air velocity and the volumetric flux, which are usually brought out as their interpretation.

For bubbly flow the velocity obtained by noise correlation corresponds with the theoretically derived velocity of bubbles without wall interaction. For more understanding of the measured velocity a detailed knowledge of the bubble and velocity profile over the tube cross-section is needed. Research is continued in that direction. In case of slug flow it was indicated by an independent experimental method that the measured velocity is the average velocity of developed and undeveloped slugs.

In order to gain insight into the coolant flow through a reactor core one needs to know not only the steam velocity but also the distribution of the steam - that is the flow type. The flow type at issue in this air-water loop could be determined using the probability density function of the photo-transistor signals; application in a BWR, however, needs more study.

Acknowledgement

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INSTRUMENTATION METHOD AND ANALYSIS OF SMALL TEMPERATURE FLUCTUATION IN INCORE COOLANT

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Abstract - The aim of this paper is to review a basic detection method which concerns with temperature fluctuation in upper core region. The measurement of small temperature fluctuation, which may be masked by bulk temperature variation, was carried out by obtaining two small frequency signal components using one thermocouple and a band pass filter systems after eliminating a large signal component. From the signal processing, typical PSD pattern changes were observed according to temperature fluctuation of the coolant. Obtained experimental results are discussed by comparing with theoretical estimation.

1. INTRODUCTION

It is important to detect the local coolant boiling due to excess heat flux of a fuel pin in a PWR or LMFB fuel assembly. The detection of boiling has been tried by neutronic measurement, acoustic measurement and temperature measurement. Each of these methods has advantages and disadvantages. Measurement of boiling noise by neutronic or acoustic method have advantage that boiling can be detected from outside of a reactor vessel, but it is difficult to locate the position of the local boiling fuel pin inside a reactor core. Difficulties are also associated with these two methods when boiling level is low.

Temperature measurement is more advantageous than those two methods. This method can identify the location of the local boiling, for each thermocouple is to be mounted at the outlet of a fuel subassembly. When a fuel pin being locally heated, vapor bubbles are generated from the fuel pin surface. The vapor bubbles may remain if the coolant is at the saturated temperature or may be subsequently collapsed if the coolant is subcooled. In those cases, very small part of the coolant may be locally influenced by bubbles generated.

In any cases temperature fluctuation is very small and we presented a method of detecting such fluctuation using two thermocouples having different time constants at the former SMORN I'm meeting. This time we improved the method and made a simple measuring system to detect small temperature fluctuation by using one thermocouple. In this paper, we deal with mainly experimental results and analysis with this measuring system.

2. THEORETICAL PREDICTION OF TEMPERATURE FLUCTUATION IN A COOLANT CHANNEL OF A FUEL ASSEMBLY
To estimate temperature fluctuation of coolant in the upper prenum zone above fuel assemblies is not so easy. Hishida reported (Hishida,1980) that prediction within a coolant channel is possible based on the following method.

When fuel pins are so arranged as shown in Fig.1 and with coolant flow of velocity being \( u(r, \phi) = u_0 \), the heat conduction equation for the coolant flow past abnormally heated fuel pin A, may be expressed in the cylindrical coordinate system as follow:

\[
\rho C_p u_0 \frac{\partial T}{\partial t}(r, \phi, z) = k_c \left[ \frac{1}{r} \frac{\partial}{\partial r} \left( r \frac{\partial T}{\partial r} \right)(r, \phi, z) \right] + \frac{1}{r^2} \frac{\partial^2 T}{\partial \phi^2}(r, \phi, z) + \frac{\partial T}{\partial z}(r, \phi, z) \]

\( (1) \)

where; \( T(r, \phi, z) \) is the coolant temperature at \( (r, \phi, z) \), \( \rho \) is the density of coolant, \( C_p \) is the specific heat of coolant, \( u \) is the bulk velocity of coolant and \( k_c \) is defined in terms of the molecular conductivity \( k_0 \) and the turbulent diffusivity of heat \( \varepsilon \) of the coolant as expressed by the followings:

\[
k_c = k_0 + \rho C_p \varepsilon
\]

\( (2) \)

and

\[
\varepsilon = k_c / k_0
\]

\( (3) \)

Since \( T(r, \phi, z) \) of eq.(1) is subject to the following boundary conditions; (1) in a subassembly of regular square configuration surrounding fuel pin A,

\[
\left. \frac{\partial T}{\partial r}(r, \phi, z) \right|_{\phi = 0, \pi/4} = 0
\]

\( (4) \)

(2) the amount of heat to be transferred from the running fluid to the surface of a dummy fuel pin is negligible, (3) at a sufficient radial distance \( R \) from fuel pin A, (4) the coolant temperature distribution should be finite and uniform \( z \rightarrow \infty \).

then the solution of eq.(1) can be written in the form of

\[
T(r, \phi, z) = \sum_{m=0}^{\infty} \sum_{n=0}^{\infty} U_{m,n}(r, \phi) v_{m,n}(z)
\]

\( (5) \)

\[
v_{m,n} = c_{m,n} \exp(-\beta_{m,n} z)
\]

\( (6) \)

\( \beta_{m,n} \) is defined such that

\[
\beta_{m,n} = \left\{ \frac{a^2}{A_{m,n}} \left( \frac{\rho C_p u_0}{2 k_c} \right)^2 \right\}^{1/2} - \frac{\rho C_p u_0}{2 k_c}
\]

\( (7) \)
where \( \alpha_{m,n} \) is the \( n \)-th eigenvalue corresponding to the \( m \)-th characteristic equation \( \beta_{m,n} \) to be derived from boundary conditions. Expression (5) gives the temperature profile in the stationary turbulent axial flow.

So there is possibility to detect coolant in locally higher temperature with respect to the bulk coolant temperature around an abnormally heated fuel pin. However, this higher temperature coolant flow fluctuates according to the coolant mixing due to grids and this fluctuation frequency is expected to be not so high.

With proceeding of excess heat generation in a fuel pin, boiling is generated around and heat pulse due to bubbles is superposed onto the coolant temperature expressed by expression (5).

In a previous paper (Iida, 1985), we showed the temperature distribution \( V(x,t) \) in a thermocouple due to heat pulse having duration time \( \Delta t_1 \) and interval time \( \Delta t_2 \), whose pulse height is \( E_1 \), as follows

\[
V(x,t) = \frac{2}{\pi} \sum_{n=1}^{\infty} \exp\left(-\frac{k n^2 \pi^2 t}{l^2}\right) \frac{n \pi x}{l} \left\{ f(x) \sin \frac{n \pi x}{l} \right\} 
\]

\[
+ \frac{n k \bar{u}}{l} \int_0^1 \exp\left[-\frac{k n^2 \pi^2 \lambda}{l^2} \left(E(\lambda) - (-1)^n E(\lambda')\right)\right] d\lambda
\]

(8)

where \( x \) is intermediate point between the surface (\( x=0 \)) and the center (\( x=1 \)) of a thermocouple, \( l \) is radius of a thermocouple, \( k = K/c_p \rho \) with \( K \) being thermal conductivity of the insulator of a thermocouple, \( c_p \) is specific heat and \( \rho \) is density of insulator material.

Expression (5) and eq.(8) may give the local temperature fluctuation in the coolant past the abnormally heated pin. In the measurement, output signal from a thermocouple is to be taken out in a combination form of expressions (5) and eq.(8). However, their signal levels are expected to be very low.

3. EXPERIMENTAL PROCEDURES

Though the coolant temperature involved fluctuation, we assumed that inside a thermocouple there exist two kinds of signal components. One of them corresponds to the mean value of the environmental temperature and its frequency component is low, though signal level is high. The other is high frequency component following to the environmental fast temperature fluctuation due to boiling flow mixing as expressed by expression (5) and eq.(8), though signal level is very low and is superposed on to the high signal level corresponding to the time averaged local temperature.

Usually this high level signal masks low level signal and the latter is neglected in measuring coolant temperature. Here we have to abstract these low level signals.

To eliminate the high signal level (DC-offset), we made a bridge circuit with a variable resistor and a thermocouple. The resistor was controlled automatically to assure DC-offset. In this system we set time constant to be 0.1 sec. By this way, small temperature fluctuation signal around thermocouple was abstracted in the maximum range of \( \pm 2.5 \mu V \). These values correspond to the maximum fluctuation temperature ranges of \( \pm 0.06^\circ C \) when the thermocouple is in the working temperature range of \( 100^\circ C \).

However, since these signal levels were too low to have direct signal processing, it was needed to amplify them.

To these amplified signals, such a filter system was used as follows: two
frequencies were selected which consisted of the high pass (HP) and low pass (LP) frequencies corresponding to the temperature expressed by expression (5) and this fluctuation expressed by eq.(8). For selecting two frequencies a band-pass filter was employed and then Power Spectral Densities (PSD) of signals were analysed. In this experiment, frequency regions were selected to be of 0.01 ~ 1.00 Hz for HP and of 2.00~10.00 Hz to LP.

In Fig.2 shows a block diagram of measuring systems. A model of a nuclear fuel subassembly, composed of 4×4 stainless steel rods in square array with each dimension of 10×300 and with pin gap of 10 mm, was put into a vessel having dimensions of 200×500. One of these stainless rods was an electric heater to simulate an abnormal fuel pin.

The coolant flow rate into the fuel pin array was about 10 l/min. The coolant temperature was kept at about 90°C during this experiments.

4. EXPERIMENTAL RESULTS AND DISCUSSION

Experiments were carried out using an unground type sheathed thermocouple having time constant of 0.4 sec. To have one PSD figure, 8 samples of temperature fluctuation were analysed and accumulated with one another.

Temperature fluctuations were measured at the vicinity heater top and at the position above 10 cm from the heater, respectively. In each place measurements were carried out in cases of (1) before bubble generation, (2) small bubble generation and (3) violent bubbly flow.

At the head of the heater, bubbles reach to the thermocouple before being collapsed, and characteristic PSD changes were obtained. The maximum PSD peak value shifted to the higher frequency side with increasing in bubbly flow rate. These frequency shifts are shown in Fig.3 which shows transitional relation between bubbling and violent bubbly flow.

From this figure it is seen that appearance of the maximum frequency is depending on the filter cut-off frequency change.

So, it was important to select cut-off frequencies of LP and HP filters. When these two frequencies are too closed or separated, the typical PSD change did not appear. The optimum frequency values can be anticipated from experiment and theoretical estimation using eq.(8). In our case, Fig.2 Schematic block diagram of an electronic measuring systems for temperature fluctuation using one thermocouple.

Fig.3 Change of maximum peak value of PSD dependig on LP/HP (LP=A,B,...,E). Figures represent in case of A=2.0Hz, B=4.0Hz, C=6.0Hz, D=8.0Hz and E=10.0Hz. These figures show frequency shift and amplitude changes of the peak of PSD before bubbling and violent bubbling.
the optimum value of LP/HP was 2.00/0.01 to 4.00/0.01 Hz. At the 10cm above position of the heater, small bubbles near the heater were almost collapsed, but in case of violent bubbly flow, many bubbles reached to the thermocouple.

Signals from small bubbly flow were weak though it was also still possible to detect characteristic PSD. Frequency shifts of PSD can also be seen as of Fig.3. The presence of bubble, which may or may not reach the thermocouple, apparent PSD change was obtained as seen in Fig. 4.

The sensitivity of this 1-thermocouple (TC) method was compared with 2-TC method which was reported at SMORN IV. The comparison is shown in Fig.5. As seen from the figure, sensitivity of 1-TC was lower than 2-TC method.

To this reason, it is assumed that, to the bridge circuit, which works to give DC-offset voltage to the thermocouple automatically, large electric noises from the thermocouple was mixed to the thermal emf. For the noise levels are larger than the signal levels after DC-offset and time constants are quite different from the set value. However, usefulness of this 1-TC method was much prospective than 2-TC method.

In conclusion
1. Using one thermocouple, a monitoring method of the coolant temperature fluctuation was verified.
2. The shift of maximum value of PSD was seen even at the position above the heater pin as well as at the vicinity of the heater.
3. To have more sensitive measurement, electrical noise protection is important.

Fig.4 PSD figures of the 10cm above position of the heater in the frequency range of LP/HP=8.00/0.01 Hz. 1) violent bubbling, 2) Before bubbling and 3) back ground noise.

Fig.5 Comparison of sensitivity between 2-TC method (1)) and 1-TC method (2)).
The author would like to thanks Dr. Hishida of Mitsubishi Atomic Power Industries, Inc., for valuable discussion and Mrs. Shimada, Itoh and Kumagai of Mitsubishi Electric Corporation who supported to prepare the electric bridge circuit.

REFERENCES

SAFETY RELATED APPLICATIONS
(PART I)

Session chairman: P. Bernard (France)
SUMMARY OF SESSION

Five papers have been presented in this session. Three of them concerned on-site measurement system characteristics assessment, and monitoring and signal validation. Two other papers dealt with signal processing techniques for sodium boiling detection in LMFBRs.

A. Measurement systems characteristics assessment and monitoring and signal validation.

'Sensor response time monitoring using noise analysis', presented by Hashemian et al., described the results of the application of AR methods for evaluating response time of temperature and pressure sensors. The results of laboratory and in-plant tests show that the apparent response time was greater that the directly measured one. This must be due to the non-white noise characteristics of the input noise. Sensing line modellization is also presented for monitoring purposes.

'Health test of components in nuclear reactor instrument systems using process identification' by Bergdahl and Oguma, presented the results of tests performed at Ringhals 4. Time series analysis was applied for measuring time constants of temperature sensors, and control system filter time constants. The author concluded that filter characteristics and temperature sensors' response time can be estimated with good accuracy using this technique.

'An integrated approach for signal validation in dynamic systems' by Upadyaya et al. dealt with the combination of the Generalized Consistency Check (GCC) and the Sequential Probability Ratio Test (SPRT) for on-line sensor failure detection. Application to an aluminium rolling mill was described.

General remarks concerning these techniques of on-line signal validation and measurement systems characteristics assessment and monitoring can be drawn:

- On-line continuous monitoring allows early detection of malfunctions.
- It reduces period of tests and manpower.
- The whole measurement systems performance can be monitored (sensing line, sensor, electronics etc).
- The continuous check of signal validity increases the level of the information contained in the signal.

As a more global observation, it appears to be a trend to combine and integrate deterministic and random information included in the signals, in a coherent approach for on-line diagnosis.

B. Signal processing techniques for sodium boiling detection.

'New approaches in signal processing technique for sodium boiling noise detection' by Singh et al.

'Signal processing techniques for the acoustic detection of boiling in LMFBRs' by Arkhipov et al. presented the results of specific techniques applied on recordings from KNS experiments and from the BOR 60 reactor in USSR.

Global remarks from these papers are the following ones:

- Acoustic detection appears to be a promising technique for on-line incipient boiling detection.
- The acoustic signal from boiling is highly impulsive and the detection techniques take into account this characteristic.
- The methods allow quite fast computations, which are necessary for practical on-line applications.
SENSOR RESPONSE TIME MONITORING USING NOISE ANALYSIS

H. M. HASHEMIAN, J. A. THIE, B. R. UPADHYAYA and K. E. HOLBERT
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Abstract - Random noise techniques in nuclear power plants have been developed for system surveillance and for analysis of reactor core dynamics. The noise signals also contain information about sensor dynamics, and this can be extracted using frequency, amplitude and time domain analyses. Even though noise analysis has been used for sensor response time testing in some nuclear power plants, an adequate validation of this method has never been carried out. This paper presents the results of limited work recently performed to examine the validity of the noise analysis for sensor response time testing in nuclear power plants. The conclusion is that noise analysis has the potential for detecting gross changes in sensor response but it cannot be used for reliable measurement of response time until more laboratory and field experience is accumulated. The method is more advantageous for testing pressure sensors than it is for temperature sensors. This is because: 1) for temperature sensors, a method called Loop Current Step Response test is available which is quantitatively more exact than noise analysis, 2) no method currently exists for on-line testing of pressure transmitters other than the Power-Interrupt test which is applicable only to force balance pressure transmitters, and 3) pressure sensor response time is affected by sensing line degradation which is inherently taken into account by testing with noise analysis.

1. INTRODUCTION

The safety system in nuclear power plants relies on information from temperature, pressure, and neutron sensors. The information must be accurate and, in case of a transient, must be received quickly to initiate safety system action if necessary. The accuracy is insured by periodic calibration and the speed of information flow is determined by response time testing of the sensors and other equipment in the instrument channel. The accuracy and response time are generally treated independently. This paper is concerned with the response time testing question in U.S. nuclear power plants. The sensors of main concern are temperature and pressure (pressure, level, and flow) sensors. There is some interest in response time testing of neutron sensors but very little actual in-plant testing has been performed even though a method was recently developed for direct testing of neutron sensors.\(^1\)

Most of the sensor response time tests in the U.S. plants are performed on safety system temperature and pressure sensors. About 70 percent of all operating U.S. PWRs perform periodic response time testing on their temperature and pressure sensors. In BWRs the main interest is on response time of pressure transmitters and only a few plants perform testing on some of their temperature sensors. Almost all the current testing is performed using methods other than noise analysis.

The limits for response time of safety system sensors are usually specified in the plant technical specifications. These limits are tied to the plant safety analysis calculations and must be maintained at a predetermined level as the plant is operating. Since sensor response time degradations are known to occur,\(^2\) periodic tests are performed to determine whether the sensors have response times which are less than or equal to the values assumed in the plant safety analysis. The testing interval must logically depend on the rate of sensor response time degradation. However, the rates of response time degradation of most process sensors are not currently known with enough confidence. Therefore, the testing intervals are often
selected to coincide with refueling intervals but a few plants have performed testing on a more frequent basis.

2. TESTING METHODS

The response time of a process sensor can be measured in a laboratory by exposing the sensor to a step, ramp, or sinusoidal input. When the sensor is installed in the process, response time testing can be done by a perturbation of the monitored variable. Obviously, this is not practical in nuclear power plants nor is the removal of sensors for periodic laboratory testing (especially since the response of some sensors is dependent on process conditions and installation). Therefore, methods which permit testing on installed sensors as the plant is operating (in-situ testing) are highly desirable. The noise analysis method is an obvious choice if it can be shown that its results are accurate. To date, very little work has been done to check the validity of noise analysis for sensor response time testing. In theory, noise analysis must be able to give an accurate measure of a sensor response time if the excitation noise has the required characteristics. That is, if the input noise is stationary and white, an appropriate analysis of sensor output must give a reliable estimate of the sensor response time. In practice, noise analyses have often failed to give consistent response time results when just one number, an apparent response time, is used to characterize a complex dynamic system. The main challenge is that the characteristics of the input noise cannot be readily verified with the existing array of sensors in most power plants. In addition, the results of noise analysis are often very sensitive to analysis parameters such as the autoregression model order used in fitting of the data, and special effort must be made to obtain the appropriate model order for a unique result. These considerations have limited the use of noise analysis to degradation monitoring rather than a method for measuring a sensor response time. It must be pointed out also that noise analysis is useful for determining the response of the sensors to small process transients rather than large transients which may be encountered in nuclear power plants. If the sensor behavior is known to be linear, the response to small transients will give comparable results to those of the large process transients.

Before 1975, noise analysis was the only known method for in-situ response time testing of process sensors. Since 1975, other methods have been developed, validated, and used. The new methods have been proven to be quantitatively more exact than noise analysis but they are not as versatile. The most advanced technique is the Loop Current Step Response (LCSR) test. The LCSR test is applicable to resistance temperature detectors (RTDs) and thermocouples. (3) It is based on remote heating of the sensor with an electric current applied through the sensor leads. The current causes a temperature transient in the sensor that can be analyzed to give its response time under the conditions tested. For pressure transmitters, a new method called power interrupt (PI) test has been developed. (4) The PI test is performed by turning the power to the transmitter off and then on. When the power is turned on the transmitter output is monitored and analyzed to give the sensor response time. This method is applicable to force-balance pressure transmitters of the type manufactured by Foxboro company. For other type of pressure sensors, a method for testing an installed sensor is available but requires access to the sensor and can therefore be performed only when the plant is at cold shutdown. The method involves using a hydraulic pressure ramp generator to provide a test signal to the sensor and simultaneously to a fast-response reference sensor. The delay time between the response of the test sensor and the reference sensor when they reach a set-point is called the sensor response time. The method is called the substitute process variable test or ramp test. There has also been some limited use of rapidly repositioning sensing line valves to introduce pressure transients to permit deducing response times.

The LCSR, PI, ramp, and pressure transient test methods are usually referred to as direct tests because they provide the actual response time of the sensors as opposed to indirect tests such as noise analysis which provide an index other than the response time - called apparent response time here - that has a known relationship to the response time.

In spite of the availability of direct tests for sensor response time testing, noise analysis remains a viable option and more validation work is justified and should be done. This is because noise analysis, if proven valid, is an excellent tool for continuous on-line testing of many types of sensors. The noise analysis technique is especially useful for response time testing of pressure sensors because, unlike other methods, it also tests the sensing line. Furthermore, except for the force-balance sensors, there is no method available to test pressure sensors while the plant is operating. A case study is included here to demonstrate the benefits of noise analysis for testing of pressure sensors.

The work presented herein is a small initial attempt into examination of noise analysis for sensor response time testing in nuclear power plants.
3. ANALYSIS TECHNIQUES

The methods for analysis of sensor noise data are extensively covered in readily available literature including previous SMORN conference proceedings and in other papers in this SMORN V conference. Therefore, an elaborate discussion on analysis techniques is not presented here. Instead only a short description of frequency and time domain methods is given below.

In the frequency domain, a direct Fourier transform of sensor output gives the power spectral density (PSD) of the signal which can be used to estimate the sensor response time. If the sensor is assumed to be a first order system excited by white noise, the reciprocal of break frequency in the power spectrum gives an estimate for its response time. For higher order dynamics, the response time can be estimated by a fit of power spectrum to an analytical expression for a noise source and transfer function which best represents the noise excitation and the sensor.

In the time domain, the sensor output $x(t)$ is represented by an autoregressive (AR) model:

$$x(t) = \sum_{i=1}^{n} a_i x(t - i\Delta t) + w(t)$$  \hspace{1cm} (1)

Assuming that the input noise, $w(t)$, is stationary and white, a fit of sensor output data to Eq. (1) gives the parameters $\{a_i\}$ which are then used to obtain the impulse response $x_I(k)$:

$$x_I(k) = \sum_{i=1}^{n} a_i x_I(k-i)$$  \hspace{1cm} (2)

The step response, $x_S(t)$, is:

$$x_S(t) = \int_{0}^{t} x_I(r)dr$$  \hspace{1cm} (3)

For temperature sensors, the response time is described in terms of an overall time constant which can be obtained directly from the step response curve. The time constant is defined as the time at which the step response attains 63.2% of its steady state value.

For pressure sensors, the response time is usually described in terms of a ramp time delay (Figure 1). This is because design basis accidents result in pressure transients that approximate a ramp. The ramp response $x_R(t)$ is computed from the step response (Eq. 3):

$$x_R(t) = \int_{0}^{t} x_S(r)dr$$  \hspace{1cm} (4)

The ramp time delay is the lag between the ramp response and the input ramp as illustrated in Figure 1. The power spectrum $S_{XX}(f)$ of the noise signal can also be calculated from the AR model as:

$$S_{XX}(f) = \sigma^2 \Delta t / \left| \sum_{k=1}^{n} a_k \exp(-j2\pi k\Delta t) \right|^2$$  \hspace{1cm} (5)

where $\Delta t$ is the sampling interval and $f$ is the frequency in Hz $[0 \leq f \leq 1/(2\Delta t)]$. 
Fig. 1. Response of a typical sensor to a ramp input.

4. SENSOR RESPONSE TIME RESULTS FROM NOISE ANALYSIS

This section presents the results of laboratory and field testing performed to examine the ability of noise analysis for providing the response time of typical process sensors.

4.1 Laboratory testing

Laboratory testing was performed on a few temperature and pressure sensors of the type used in U.S. nuclear power plants. A table-top hydraulic pressure test loop was constructed to provide pressure noise data. The loop consisted of a fast-response reference sensor and two Foxboro pressure sensors. The reference sensor is used to provide the characteristics of the input noise.

The ramp time delays of the two Foxboro pressure transmitters were first measured directly using the test method mentioned in Section 2. The results of this test is referred to as direct test results. The sensors were then tested with noise analysis. The results are shown in Table 1. The PSDs for the two pressure transmitters are shown in Figure 2 and 3. A discussion of the peaks in these PSDs is given in the next section.

Table 1. Results of laboratory testing of pressure sensors.

<table>
<thead>
<tr>
<th>Sensor Type</th>
<th>Apparent Response Time</th>
</tr>
</thead>
<tbody>
<tr>
<td></td>
<td>Noise Analysis (sec.)</td>
</tr>
<tr>
<td>Pressure Sensors</td>
<td></td>
</tr>
<tr>
<td>Foxboro #1</td>
<td>0.84</td>
</tr>
<tr>
<td>Foxboro #2</td>
<td>0.73</td>
</tr>
<tr>
<td>Temperature Sensors</td>
<td></td>
</tr>
<tr>
<td>RTD</td>
<td>1.44</td>
</tr>
<tr>
<td>Bare Thermocouple</td>
<td>0.14</td>
</tr>
</tbody>
</table>

Laboratory noise tests were also performed on an RTD and a general purpose bare thermocouple. The test set-up consisted of a rotating tank with room temperature water along with a hot/cold water injection system. Temperature noise is produced by random injection of
hot/cold water into the flowing room temperature water in the tank. The test results (sensor overall time constants) are included in Table 1. The sensor PSD is shown in Figure 4. The direct test results for the temperature sensors are from sudden immersion of the sensors into the rotating tank water at the same flow rate as was used during the noise test.

The results in Table 1 indicate that noise analysis produces larger values for response times than the direct tests. This is consistent with results of earlier work performed by other authors. One obvious cause for longer values can be the bandwidth of the noise sources. We can examine this cause for the temperature sensors here. The response time of the thermocouple from noise analysis is 0.14 second which corresponds to a break frequency of about 1.1 Hz. Since the thermocouple and the RTD were tested with the same input noise, we can say that the bandwidth of the driving noise for the RTD was at least 1.1 Hz. This bandwidth is adequate for testing the RTD and must have produced better results if the excitation noise was primarily responsible for the accuracy of the results. This argument and other observations during this and other projects have led to the conclusion that the actual response time of sensors cannot be expected to be the same as the apparent response time obtained by noise analysis but the noise analysis results are useful for comparative
evaluation of a group of sensors, for determining relative response time values, and for detecting if these values change as the sensors age in a process. It is evident from the values in Table 1 that noise analysis results are proportional to direct (actual) response times but are always larger. This is important in nuclear power plant applications where test methods must provide conservative results if they are unable to produce the true results.

4.2 In-Plant testing

Noise tests were performed on temperature and pressure sensors in operating PWRs. The noise tests were performed in conjunction with LCSR testing for RTDs and PI testing of pressure transmitters. The results are given in Table 2 and compared with those obtained by direct testing of the same sensors using the exact methods (the RTD test results are from an earlier work described in Reference 6). Note that the noise analysis results are again conservative as they were in the case of the laboratory results.

Table 2. Test results for RTDs and pressure sensors in operating PWRs.

<table>
<thead>
<tr>
<th>Sensor</th>
<th>Apparent Response Time</th>
<th>Response Time</th>
</tr>
</thead>
<tbody>
<tr>
<td></td>
<td>Noise Analysis (sec.)</td>
<td>Direct Test (sec.)</td>
</tr>
<tr>
<td>RTDs</td>
<td></td>
<td></td>
</tr>
<tr>
<td>RTD #1</td>
<td>8.2</td>
<td>3.5</td>
</tr>
<tr>
<td>RTD #2</td>
<td>6.9</td>
<td>4.9</td>
</tr>
<tr>
<td>RTD #3</td>
<td>8.5</td>
<td>5.8</td>
</tr>
<tr>
<td>Pressure Sensors</td>
<td></td>
<td></td>
</tr>
<tr>
<td>Pressure Sensor #1</td>
<td>0.31</td>
<td>0.11</td>
</tr>
<tr>
<td>Pressure Sensor #2</td>
<td>0.38</td>
<td>0.10</td>
</tr>
<tr>
<td>Pressure Sensor #3</td>
<td>0.39</td>
<td>0.14</td>
</tr>
</tbody>
</table>

In some cases the power spectrum of reactor noise data has very poor broadband behavior. An example is shown in Figure 5 where the PSD has no break through the low and high frequency range. This is from a cold leg RTD in a PWR. In such cases the noise analysis method will not provide any useful information about the sensor response characteristics.
Fig. 5. Power spectrum of PWR cold leg temperature noise signal.

5. MODELING OF PRESSURE NOISE

A simple physical model for the table-top pressure loop (Figure 6) was developed and used to determine the validity of the PSD results obtained for the two Foxboro transmitters and to quantify faults that the PSD can detect. This involved calculating the PSDs and comparing them with those obtained from analysis of actual noise data sampled in the laboratory for the two Foxboro transmitters. This physical modeling approach is very useful if one is to quantitatively identify the components which combine to make up the apparent response time.

Fig. 6. Schematic of the laboratory pressure loop.

Among fault types that influence pressure sensing system PSD's are diaphragm stiffness changes and sensing line resistance changes. The parameter-fitted model is used to demonstrate the basic applicability of noise analysis for identifying gross calibration changes in pressure transmitters and for determining gross blockages in the sensing lines. The calibration check involves using noise data to identify changes in diaphragm stiffness in a pressure transmitter and the determination of sensing line blockages involves monitoring the changes in the values of sensing line resistance. A more elaborate discussion on sensing line testing with noise analysis is given in Reference 7.

5.1 Impedance Model

The dynamics of pressure phenomena in pipes are analogous to electrical voltage phenomena in equivalent circuits. The analogies are:
voltage \propto \text{pressure}
current \propto \text{volume displacement}
resistance \propto \text{resistance}
inductance \propto \text{inertia}
capacitance \propto \text{capacity}

The last two quantities may be computed from quantities responsible for inertia and springiness:

- \text{inertia} = \frac{M}{\rho \ell} \quad \text{density x length/area (for a fluid in a sensing line)}
- \text{inertance} = 0 \quad \text{(for a transmitter diaphragm)}
- \text{capacitance} = \frac{C}{\rho \ell} \quad \text{length x area/(density x c²) (for a fluid in a sensing line)}
- \text{capacitance} = \frac{V}{P} \quad \text{volume displaced per unit pressure (for a transmitter diaphragm)}

where \( c \) is the velocity of sound. Figure 7 shows the circuit diagram corresponding to the piping structure of the laboratory pressure loop. In this figure, the capacitor \( C_2 \) represents the Foxboro transmitter which is closer to the noise source (sensor #2) and \( C_1 \) represents the one which is farther away from the noise source (sensor #1). The inertance of the main pipe is \( M_0 \). The two Foxboro pressure transmitters in the test loop are represented by the capacitances \( C_1 \) and \( C_2 \). The inertances of the pipes leading to the two transmitters are \( M_2 \) and \( M_1 \) for the short and the long sensing lines leading from the noise source to the two transmitters. \( R_1 \) represents the resistance of the long sensing line plus the resistance due to any partial blockage in that line. Additional capacitances due to sensing line volumes are not shown because their calculated values indicated that their effects are negligible at the low frequency of interest in this study.

![Circuit diagram](image)

**Fig. 7.** Circuit diagram corresponding to piping structure of laboratory pressure loop.

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5.2 Validation of the PSD Results

The acoustic impedance model of the pressure loop was used to calculate the PSDs of the two Foxboro transmitters in the loop. The calculated PSDs were then compared with the PSDs obtained by processing of pressure noise data from each transmitter. This showed good agreement between the calculated and measured PSDs in the frequency range of interest (see Figures 8 and 9). Figure 8 shows the fit of impedance model's calculated PSD to the measured PSD for sensor #2 whose short sensing line and sensor capacitance is responsible for the 8.2 Hz peak. Figure 9 shows the fit of impedance model's calculated PSD to the measured PSD for sensor #1 whose long sensing line and sensor capacitance is responsible for the 3.2 Hz peak.
The equation used in these fits for the PSD of the pressure noise measured from the volumetric displacement of \( C_2 \) or \( C_1 \) can be simply formulated from the impedances:

\[
\text{PSD of pressure} = \frac{F \times \text{noise source}/j\omega}{j\omega M_s + [Z_1 Z_2/(Z_1 + Z_2)]^2}
\]  

(6)

where \( Z_1 \)'s are the branch impedances of the two sensing lines. These are defined as:

\[
Z_1 = R_1 + j\omega M_1 + 1/j\omega C_1
\]  

(7)

The factor \( F \) is the branch splitting factor of the currents, namely \( Z_1/(Z_1 + Z_2) \).

The validation work involved generating the PSDs using the following procedure:

1. Determine the loop inductances, \( M_s \), from their geometric dimensions.
2. Choose \( C_1 \) values to give the experimentally observed resonant frequencies (3.2 and 8.2 Hz).
3. Select values of \( R_1 \) to obtain the best fit of the theoretical PSD shape to the data (with rms values normalized to be the same).
5.3 Quantifying Sensing Line Degradation and Sensor Calibration

The line resistances that gave the best fit were $950 \times 10^5$ and $142 \times 10^5$ lb/ft$^4$ sec. These results were obtained for the long and short sensing line respectively from the test with no degrading blockage in the long line (i.e., its blockage valve was at 90° meaning that it was fully open). With valve positions of 45° and 23° in two tests where degradation was introduced, the results of the above analysis were resistances of the long line 10% and 25% higher than for the unblocked case.

The diaphragm stiffnesses $C_1$ and $C_2$ for the two Foxboro transmitters were obtained from the two measured resonant frequencies (3.2 and 8.2 Hz) in the two transmitters measured spectra. Additional data used included $N_g$, the inerance of the main pipe, and $N_1$ and $N_2$, the inerances of the long and short sensing lines. These were identified by measuring the dimensions of the sensing lines.

Hence in fitting the shape of the two measured PSD's to the two theoretical expressions from Eq. 6, the four model parameters adjusted for a best fit are: $R_1$ and $R_2$ (whose values are given above), and $C_1$ and $C_2$. Best fit values obtained for these latter are:

$$C_1 = 2.36 \times 10^{-10} \text{ } \text{ft}^4\text{sec}^2/\text{lb},$$

$$C_2 = 3.25 \times 10^{-10} \text{ } \text{ft}^4\text{sec}^2/\text{lb}.$$ 

Since sensor calibration is influenced by these values, any sensor diaphragm stiffness degradation can be quantified by this noise analysis method. Moreover, air bubbles in the sensing line, when they significantly influencing system stiffness, can be detected by data-to-model comparisons.

6. CONCLUSIONS

Noise analysis has the potential for on-line monitoring of response time of most process sensors. However, the method relies on fundamental assumptions that cannot be verified without a comparison sensor and the accuracy of the method for sensor response time testing is not adequately known until more experience is accumulated and adequate physical modeling for interpretation is used. Presently, these considerations restrict the noise analysis to sensor response time degradation monitoring rather than a method for quantitative response time testing. The method must initially be used in conjunction with direct testing methods which have verified accuracies. Direct methods are currently available for temperature and pressure sensors. A combined use of direct methods with noise analysis and additional laboratory testing and physical modeling will build up the needed experience for using the noise analysis technique.

A fortunate observation here is that the noise analysis has almost always given conservative response time results. This is important in nuclear power plant application where test methods must produce conservative results if they cannot produce the true results.

REFERENCES

HEALTH TEST OF COMPONENTS IN NUCLEAR REACTOR INSTRUMENT SYSTEMS USING PROCESS IDENTIFICATION

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Abstract - A process identification technique using an auto regressive model has been applied to measurements at Ringhals 4, a PWR in Sweden. The purpose of the present work is to develop a method for testing the dynamics of sensors and signal conditioning circuits in safety systems as well as the regulator parameters during normal operation of the reactor. The noise analysis was performed for a coolant temperature safety system and a pressure regulator in a loop for control of boron concentration in the reactor. The results show convincingly that filter characteristics as well as temperature sensor response time can be estimated with good accuracy. Therefore the present method enables us to evaluate integral performance of instrumentation chains in a safety system during reactor operation. The same conclusion holds for the analysis of the controller. On-line application of the present method to test components in the reactor safety system would increase the system reliability. It also results in reduction of manpower by replacing the component test, performed manually, by automated on-line tests during the reactor operation.

1 INTRODUCTION

Noise analysis has been performed at Ringhals 4, a pressurized water reactor of Westinghouse type in Sweden.

An important safety circuit in the NPP has the hot leg and cold leg temperature as inputs. Via summing networks differential and mean value of the temperature signals are formed. Each of these variables are lag filtered to reduce influence of spikes. The lag filters are followed by lead-lag filters to speed up the formed signals before connection to the protection system.

Every year test of the lag and lead-lag filters is performed to make sure that no change of dynamic characteristics has occurred.

The filter units are tested with step and ramp inputs and corresponding output signals are recorded to make it possible to decide time constants for the filters. The same type of test is also performed for the controllers in the power plant to evaluate proportional band and integral action time. These tests give a good picture of the component status but they demand a lot of work during revision of the reactor.

With the view to develop an alternative method permitting evaluation of the above mentioned component tests during reactor operation, an autoregressive (AR) method was applied to the noise signals from input and output of each component.
With the identified models it is possible to estimate filter time constants and controller parameters.

In addition to this the method will also give the response time of the temperature sensors in the safety circuit.

The following components and characteristics were analysed and numerical values for estimates were obtained. A comparison was also made with values from traditional test results.

- Estimation of response time of the temperature sensors in hot and cold leg in above mentioned safety circuit.

- Calculation of time constants in lag and lead-lag filters in the same circuit.

- Estimation of proportional band and integral action time of a controller.

In estimating the response time of a temperature sensor, an autoregressive method has been applied by Upadhyaya and Kerlin 1978 and Jacquot et al., 1983. An effort has also been made by Eklund, 1984 to estimate regulator parameters based on a recursive identification. With the authors' best knowledge, however, there has been no publications which covers dynamics test of the complete instrumentation chain.

2 THE SAFETY CIRCUIT FUNCTION

The thermal power from the reactor is released via the reactor cooling water circulating from reactor vessel to the steam generator and back again. In Ringhals 4 there are three coolant loops.

The temperature in hot and cold leg are measured by RTD's (Resistance Temperature Detector). The differential temperature of hot leg and cold leg is an input to the power control system.

The RTD's are also part of the safety circuit system. Figure 1 shows the main idea of the safety circuit. A big change in ΔT and Tavg will cause shut down of the reactor. ΔT and Tavg are formed with the aid of passive summing networks. These signals are low pass filtered to prevent electrical spikes to cause shut down of the reactor. The signal conditioning unit also includes lead-lag filters to speed up the signals delayed by the low pass filters.

As a principle ΔT and Tavg passes equivalent signal conditioning units. There is, however, a numerical difference in the filter parameters τ, compare Figure 1.
PSN = Passive Summing Network
R/E = Resistance to voltage converter
TP = Test point
ΔT = Differential temperature hot leg
cold leg \( (t_1 - t_2) \)
Tavg = Mean value temperature hot leg and
cold leg \( (t_1 + t_2)/2 \)
s = Laplace operator
τ = Filter time constant

Fig. 1 The temperature sensing protection system

2.1 PROCESS IDENTIFICATION TO ESTIMATE THE TEMPERATURE SENSOR DYNAMICS

The temperature sensor dynamics can be estimated when the signals TP/422H and TP/422 are sampled by the measurement computer. The algorithm to calculate the step response is only given by a simple description here. More information is available in (1).

The sensor signal is called X(t). After sampling the continuous variable is transferred to a time series X(1), X(2), X(3), ... An Auto-Regressive (AR) model is fitted to the time series with following structure.

\[
M
X(t) = \sum_{m=1}^{M} A(m) X(t-m) + e(t)
\]  \hspace{1cm} (1)

where e(t) is white noise and
A(m) are coefficients in the transfer function

The main problem of identification is to find the coefficients A(m) and to determine the model order M which minimizes the variance of e(t).
As soon as the identification is performed the sensor time constant can be calculated by a mathematical step test of the model. The time $t$ when $Z(t)$ in equation 2 is 0.63 is the estimated time constant. A graphic presentation of the response time calculation is shown in Figure 2.

$$Z(t) = \sum_{m=1}^{M} A(m) Z(t-m) + 1.$$  \hspace{1cm} (2)

![Graph showing step test of the model.](image)

Fig. 2 Step test of the model.

2.2 RESULT OF THE SENSOR RESPONSE

The results of the response times for the temperature sensors are shown in Figures 6 and 7. Figure 6 shows the time series for cold leg and hot leg RTD and the corresponding step test of the calculated model.

The total measurement time is split into intervals M1 - M4 in Figure 7. Each interval is 5 minutes long. Sensor response time has been calculated for each interval. This method gives a simple way to find the reliability in the results. The response time for M1 - M4 only shows small fluctuations for the sensors.

It is also made clear that the cold leg sensor is some what faster in response than the hot leg sensor. It is also interesting to see that the sensor manufacturer specifications of $T=0.5$ seconds agree so well with the estimated results. Since the sensor has direct contact with the water no extra thermal delays dependent on installation are expected.

2.3 PROCESS IDENTIFICATION TO ESTIMATE THE LAG FILTER TIME CONSTANT.

The identification process is similar to what is described in Figure 3. A computer model is fed with the signals $u(t)$ and $y(t)$. A difference between $y(t)$ and corresponding $\hat{y}(t)$ causes correction of the model with the goal to reduce the difference between $y$ and $\hat{y}$. A small deviation means that there is a good agreement between model and process and that the white noise is low.
The dynamics of the low pass filter in the protection system can be estimated from a model with following structure. As soon as the coefficients $A(m)$ and $B(n)$ are known step response can be estimated with the model to calculate $\tau_1$ and $\tau_2$, see Figures 8 and 9.

The criterion used to estimate $\tau$ is the same as for the sensor model.

$$y(t) = \sum_{m=1}^{M} A(m) y(t-m) + \sum_{n=1}^{N} B(n) u(t-n) + e(t) \quad (3)$$

where
- $y(t)$ = output signal
- $u(t)$ = input signal
- $A(m)$, $B(n)$ = filter parameters
- $e(t)$ = white noise

### 2.4 RESULTS OF TIME CONSTANTS FOR THE LAG FILTERS

Results of the filter tests are given in Figures 8 and 9. The low pass filtering effect is already obvious from the input and output signals. It is easy to see that the output has lower frequency content than the input signal.

The step response character is the same as for the sensor dynamics. The agreement between model and process is also made clear from Figure 8. The continuous curve is the measurement signal $y(t)$ and the dotted curve is the model response $\hat{y}(t)$. There is a good agreement between model and process.

The time constants show small variations for the different measurement series $M1$ - $M4$. The length of each measurement in this case is 8.5 minutes. It is also made clear that the lag filter time constant $\tau_2$ is somewhat faster than $\tau_1$. The results are reasonable in comparison with traditional test results.
2.5 PROCESS IDENTIFICATION TO ESTIMATE TIME CONSTANTS OF THE LEAD-LAG FILTER

The lead-lag filter characteristics can also be estimated with the aid of process identification. Input and output signals are sampled in the same way as for the lag filters. The relation between input and output are fitted to a model with following structure.

\[
M \quad N
y(t) = \sum_{m=1}^{M} A(m) y(t-m) + \sum_{n=0}^{N} B(n) u(t-n) + e(t) \tag{4}
\]

where
- \(y(t)\) = output signal
- \(u(t)\) = input signal
- \(A(m), B(n)\) = filter parameters
- \(e(t)\) = white noise

The reader can easily see that the model is similar to the lag filter model. There is, however, an important difference since this model can handle prompt response between input and output. Thus, the summation starts with \(n=0\).

The transfer function for the analog filter is shown below. The function can also be expressed in parameters \(a\) and \(b\), where \(a\) expresses the prompt relation between input and output.

\[
\frac{1 + \tau_3 s}{K} = \frac{b}{1 + \tau_4 s} \tag{5}
\]

Theoretical step test of the filter give the following result.

\[
y(t) = \begin{cases} \frac{a}{\tau_4} & t < \tau_4 \\ \frac{a+b}{\tau_4} & t \geq \tau_4 \end{cases}
\]

where
- \(a + b = K\) \(\tag{6}\)
- \(a \tau_4 = K \tau_3\) \(\tag{7}\)

\(a, b\) and \(\tau_4\) can be extracted from the step response while \(\tau_3\) can be calculated with the equations 6 and 7 given above.

Fig. 4 Theoretical step test of a lead-lag filter.
2.6 TIME CONSTANT RESULT FOR THE LEAD-LAG FILTER

The lead-lag filters are shown in the protection circuit in Figure 1. Only one of the filters has been analysed. Figure 10 shows the input and output signals. The input signal is sampled at TP/422J and output at TP/422L, compare Figure 10. It is obvious from the figure that the frequency content is higher on output than on input. This is normal for a phase advancing network. There is a good agreement between output and estimated output from the filter. Figure 11 shows the results for different measurement series M1, M2, ... It is made clear from the results that the filter time constants can be estimated with good agreement with traditional tests.

3 ESTIMATION OF REGULATOR PARAMETERS

A pressure regulator has been analysed with the aid of process identification. The aim was to find the proportional and integral action of a PI regulator.

The circuit where the regulator is working is used to introduce boron into the reactor water. The pressure regulator reduces the pressure from 156 to 30 (bar). The reference value to the regulator is constant 30 (bar). The regulator was running in automatic mode during the measurements. The feedback pressure signal PP/145A and the actuating signal PP/145B are input and output signals to the regulator.

The mathematical model used for the regulator analysis is of the same type as the lead-lag filter model. A PI-regulator gives prompt response on inputs similar to the character of a lead-lag filter.

The following definitions are valid for a PI-regulator.

\[
\begin{align*}
\text{Reference input} & \quad \rightarrow \quad \bullet \quad \rightarrow \quad F(1 + \frac{1}{sTi}) \quad \rightarrow \quad \text{Actuating signal} \\
\text{Feedback signal} & \\
\text{Actuating signal} & \\
2FA & \quad \rightarrow \quad \text{tid} \\
FA & \quad \rightarrow \quad \text{Ti}
\end{align*}
\]

where \( F \) = gain of the regulator
\( -A \) = step of the input signal
\( Ti \) = integral action time (s)
\( P \) = proportional action
and \( P = 100 \left( \frac{1}{F} \right) \% \)

Fig. 5 Definitions for a PI-regulator.
3.1 RESULT OF PI-REGULATOR ANALYSIS

The input signal to the regulator is PP/145A and the output signal from the regulator is PP/145B. Both signals are shown in Figure 12. The figure also shows step response and the agreement between the output and estimated output signal. From the figure it is obvious that the output signal and estimated output are in good agreement. This means that the model is a good description of the process. Step test with the model gives the parameter's proportional action in (%) and the integral action time in (s). The results for different measurement series only show small fluctuations and they agree well with results from standard tests.

4 CONCLUSIONS

The results show convincingly that filter characteristics as well as temperature sensor response time can be estimated with good accuracy. Therefore the process identification method enables us to evaluate under operation integral performance of instrumentation chains in a safety system.

The same conclusion holds for the analysis of the regulator.

On-line application of the present method to component tests in the reactor safety system would contribute to increase the system reliability. It also results in reduction of man power by replacing the component test, performed manually, by automated on-line tests during the reactor operation.

REFERENCES


TEMPERATURE SENSOR SIGNALS

\[
\text{COLD LEG}
\]

\[
\text{HOT LEG}
\]

Time [sec]

COMPUTER CALCULATED RESPONSE FOR THE SENSORS

Step response of TP/422

\[
\text{COLD LEG}
\]

\[
\text{HOT LEG}
\]

Time [sec]

Fig. 6 Temperature signals from the sensors and computer calculated response.

<table>
<thead>
<tr>
<th>MEASUREMENT NO</th>
<th>( \tau(S) ) COLD LEG</th>
<th>( \tau(S) ) HOT LEG</th>
<th>( \tau(S) ) MANUFACTORY SPECIFICATION</th>
</tr>
</thead>
<tbody>
<tr>
<td>M1</td>
<td>0.336</td>
<td>0.680</td>
<td>-</td>
</tr>
<tr>
<td>M2</td>
<td>0.312</td>
<td>0.498</td>
<td>-</td>
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<tr>
<td>M3</td>
<td>0.357</td>
<td>0.459</td>
<td>-</td>
</tr>
<tr>
<td>M4</td>
<td>0.306</td>
<td>0.506</td>
<td>-</td>
</tr>
<tr>
<td>MEAN VALUE</td>
<td>0.33</td>
<td>0.54</td>
<td>0.5</td>
</tr>
</tbody>
</table>

Fig. 7 Result of response time calculation for temperature sensors in cold leg och hot leg.
INPUT SIGNAL TP/422A

OUTPUT SIGNAL TP/422J

MODEL STEP TEST FROM TP/422A TO TP/422J

OUTPUT SIGNAL Y(T) AND ESTIMATED OUTPUT SIGNAL \( \hat{Y}(T) \)

Fig. 8 Result of lag filter test

<table>
<thead>
<tr>
<th>MEASUREMENT NO</th>
<th>( \tau_1(S) ) TP/422A - TP/422J</th>
<th>( \tau_2(S) ) TP/422C - TP/422K</th>
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</thead>
<tbody>
<tr>
<td>M1</td>
<td>2.31</td>
<td>1.92</td>
</tr>
<tr>
<td>M2</td>
<td>2.49</td>
<td>1.81</td>
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<td>M3</td>
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<td>1.93</td>
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<tr>
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<td>1.88</td>
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<tr>
<td>MEAN VALUE</td>
<td>2.37</td>
<td>1.86</td>
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Fig. 9 Result of lag filter test in protection circuit 422.
Fig. 10  Result of lead-lag filter test

<table>
<thead>
<tr>
<th>MEASUREMENT NO</th>
<th>$\tau_3(S)$</th>
<th>$\tau_4(S)$</th>
</tr>
</thead>
<tbody>
<tr>
<td>M1</td>
<td>33.7</td>
<td>9.36</td>
</tr>
<tr>
<td>M2</td>
<td>31.9</td>
<td>8.97</td>
</tr>
<tr>
<td>M1 + M2</td>
<td>32.5</td>
<td>9.12</td>
</tr>
<tr>
<td>MEAN VALUE</td>
<td>32.7</td>
<td>9.15</td>
</tr>
</tbody>
</table>

Fig. 11  Result of the time constant estimations for the lead-lag filter in safety circuit 422.
Fig. 12 Result of regulator test

<table>
<thead>
<tr>
<th>MEASUREMENT NO</th>
<th>P (%)</th>
<th>TI (S)</th>
</tr>
</thead>
<tbody>
<tr>
<td>M1</td>
<td>296</td>
<td>-</td>
</tr>
<tr>
<td>M2</td>
<td>287</td>
<td>-</td>
</tr>
<tr>
<td>M3</td>
<td>284</td>
<td>-</td>
</tr>
<tr>
<td>M4</td>
<td>290</td>
<td>-</td>
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<tr>
<td>MEAN VALUE</td>
<td>289</td>
<td>304</td>
</tr>
</tbody>
</table>

Fig. 13 Result of the proportional and integral action of the regulator 145.
AN INTEGRATED APPROACH FOR SIGNAL VALIDATION IN DYNAMIC SYSTEMS

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ABSTRACT - Process variables measured by a multitude of sensors (neutron power, temperatures, pressures, flow rates, fluid levels), are used as inputs to control systems, protection systems and monitoring systems in nuclear power plants. Integration of diverse signal processing schemes will provide more complete information for signal verification and also for process diagnostics. The combination of generalized consistency check (GCC) and sequential probability ratio test (SPRT) improves the reliability of the decision/estimation procedure. This paper integrates the dynamic signal component analysis method with the GCC, and provides an improved technique for detecting and isolating sensor maloperation and process anomaly (mechanical components, controllers, etc.). This technique is also applicable to process control systems.

1. INTRODUCTION

Sensor failure detection and isolation methods have been used in aerospace and in nuclear power plants to track both rapid and slow degradations (Upadhyaya, 1985; Deckert et al., 1977; Deutch et al., 1987). Recently, these techniques are being applied more actively in the process control industry (Upadhyaya et al., 1987b). Signal validation is necessary (a) to improve the quality of the product through improved control, (b) to improve reliability of operator decision, (c) to monitor both slow and fast degradation of sensors, and (d) as an aid in scheduling maintenance, and thus minimize plant down time.

The basic principle of the procedure used in the past is to identify sensor redundancy and to combine it with analytical redundancy to provide diversity. The inconsistent measurements are identified and the corresponding sensors are isolated from the set, and if necessary the signal is estimated using the remaining measurements. This approach often provides a partial information. We have expanded the scope of the problem by processing sensor outputs using both low frequency and high frequency information reduction techniques. This integration is applicable to a variety of processes. The GCC and the modified SPRT (Chien and Adams, 1976) are developed for single channel analysis and for a network of sensors. Primarily, the DC (or low frequency) information is processed by this module. Analytical redundancy is generated either by physical models or by empirical data-driven models. A new technique using an optimal polynomial data fit has been developed at the University of Tennessee (Upadhyaya et al., 1987c) for state prediction. This method exploits the large database of plant operating conditions to learn the model characterization. The higher frequency analysis makes use of multivariate autoregressive (MAR) modeling technique, and a systematic procedure to interpret the cause-and-effect signatures derived from the model. Development of a simple expert system is underway to perform the analysis for detecting and isolating sensor maloperation and process anomaly (mechanical components, controllers, etc.).

In Sect. 2 the generalized consistency checking (GCC) and the modified sequential probability ratio test (MSPRT) are described. Application of this method for detecting bias and noise-type degradations is discussed in Sect. 3. The application of data-driven modeling (multivariate autoregressive modeling) to the Aluminum Company of America's rolling mill facility in Alcoa, Tennessee is described in Sect. 4. Even though this application is not related directly to nuclear power plants, the feasibility of applying multivariate noise analysis for rotating

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system diagnostics is demonstrated for a real process. Summary and concluding remarks are given in Sect. 5.

2. COMBINED GCC AND MSVRT APPROACH FOR SIGNAL VALIDATION

2.1 The Generalized Consistency Checking (GCC) Algorithm

The GCC algorithm (also called the parity-space method (Potter and Suman, 1977)) is a systematic technique of comparing a set of redundant measurements to detect and isolate faulty signals. Both direct measurements and analytically generated measurements are used to perform this evaluation. Figure 1 shows a schematic of this procedure with the "OR" gate indicating the decision/estimator (D/E) module, and the "AND" gate signifying the use of all pertinent information in generating analytical measurements. Properly tuned analytical or empirical models are necessary for the latter. Efforts are currently underway at the University of Tennessee (Upadhyaya et al., 1987) to develop optimal steady-state and quasi steady-state polynomial state prediction models using normal plant operation database.

The basic GCC algorithm is similar to that developed by Desai and Ray (1981). If \( m_i \) and \( m_j \) are two measurements of the same process variable, and if \( \eta_i \) and \( \eta_j \) are the respective allowable tolerances, then \( m_i \) and \( m_j \) are said to be pairwise inconsistent if

\[
|m_i - m_j| > \eta_i + \eta_j \quad \text{for all } i \neq j .
\]  

(1)

For a set of \( t \) signals, after checking \( t(t-1)/2 \) pairs, an inconsistent index \( I_t \) is assigned to the measurement \( m_i \). \( I_t \) is increased (decreased) as the inconsistency increases (decreases), and varies between zero and \( t-1 \). The further management and isolation of faulty readings is based on the following: the maximum \( I_{\text{max}} \) and minimum \( I_{\text{min}} \) error indices, the number of measurements, \( N_{\text{max}} \) and \( N_{\text{min}} \), having the maximum and minimum error indices, respectively. A complex logic procedure has been developed to perform detection and isolation based not only on the current information but past values of the variable estimate and consistency of signals.

When \( (t-k) \) measurements are excluded from the set, the remaining \( k \) measurements are used to obtain a state estimate and is given by

\[
x^*(t) = \frac{\sum_{i=1}^{k} w_i (t-I_i) m_i(t)}{\sum_{i=1}^{k} w_i (t-I_i)}
\]

(2)

Each weight \( w_i \) is inversely proportional to the inconsistency index, \( I_i \). Several options are built into the new GCC algorithm, with possible feedback from the SPRT submodule. The algorithms also indicate whether the degraded sensors are reading "high" or "low" compared with the estimate \( x^* \).

2.2 The Modified Sequential Probability Ratio Test (MSVRT)

The GCC algorithm performs signal validation based on sample by sample evaluation. The sequential probability ratio test (SPRT), first developed by Wald (1947), is applied to every signal by computing the residual between \( x^*(t) \) and \( m_i(t) \). This is a cumulative error index and avoids false alarms due to occasional spurious signal changes. Since this is applied to individual signals, the behavior of the cumulative index is useful in identifying common-cause failure and for detecting incipient failure. The estimate \( x^*(t) \) is obtained from the GCC, thus an error in GCC may increase the false alarm rate of SPRT.

We define the residual between the measurement \( m(t) \) and the estimate \( x^*(t) \) at sample instant \( k \) as

\[
S_k = m(k) - x^*(k)
\]

(3)

The residuals each have a Gaussian distribution with mean \( \mu_0 \) and variance \( \sigma_0^2 \) during normal mode

\[
p_0 = p(S_k; \mu_0, \sigma_0^2) = \frac{1}{\sqrt{2\pi\sigma_0^2}} \exp \left\{ -\frac{(S_k-\mu_0)^2}{2\sigma_0^2} \right\} .
\]

(4)

The degraded sensor residual density \( p_1 = p(S_k; \mu_1, \sigma_1^2) \) is defined to have a Gaussian density.
The basis of Wald's SPRT is to compute the log likelihood ratio (LLR) as

\[ \lambda_n = \ln \frac{p(S_1, S_2, \ldots, S_n | \mu_1, \sigma_1^2)}{p(S_1, S_2, \ldots, S_n | \mu_0, \sigma_0^2)} \quad (5) \]

For the Gaussian case with independent samples \( S_i \) this can be updated recursively in the form

\[ \lambda_n = \lambda_{n-1} + \ln \frac{p(S_n | \mu_1, \sigma_1^2)}{p(S_n | \mu_0, \sigma_0^2)} \quad (6) \]

Wald's sequential test states that, at any stage \( n \), if \( \lambda_n > B \) accept the failed mode, if \( \lambda_n < A \) accept the normal mode, and if \( A < \lambda_n < B \), continue the test. The bounds \( A \) and \( B \) are set using the false alarm probability (\( \alpha \)) and the missed-alarm probability (\( \beta \)) and are given by (Wald, 1947)

\[ A = \ln \left( \frac{\beta}{1-\alpha} \right), \quad B = \ln \left( \frac{1-\beta}{\alpha} \right); \quad 0 < \alpha, \beta < 1. \quad (7) \]

For the detection of noise-type degradation, \( \nu_0 = \nu_1 = 0 \) and

\[ \lambda_n = \lambda_{n-1} + \frac{1}{2} \left[ \frac{1}{\sigma_0^2} - \frac{1}{\sigma_1^2} \right] S_n^2 + \ln \left[ \frac{\sigma_0}{\sigma_1} \right]. \quad (8) \]

The SPRT checks for both normal and failed states. A modification of this was developed by Chien and Adams (1976) for the bias degradation. Hereafter referred to as the MSFRT, the measurement is assumed normal and the test is made to detect only the failed state. A new upper bound, \( B^* \), was derived using the mean time, \( T \), between false alarms and tolerances on mean bias and signal variance. The relationship between \( B^* \) and \( T \) has the form

\[ T = \frac{2\sigma^2}{\nu_1^2} (e^{B^*} - B^* - 1) \quad (10) \]

An approximation for large \( T \) (useful in applications) is

\[ B^* = \ln \left( \frac{\nu_1^2}{2\sigma^2} T \right) \quad (11) \]

where \( T \) represents the number of samples. The log likelihood ratio is set to zero, whenever it is negative. This control action minimizes the time of detecting the degraded state. The missed alarm probability is not defined. If a sensor degradation occurs, the probability of its detection is one. The general algorithm can be used without the interaction between GCC and SPRT, or can be implemented as a combined procedure.

3. APPLICATION TO SENSOR FAILURE DETECTION

The signal validation methodology presented in Sect. 2 was applied to experimental data from a hot air loop and detection of bias, noise, and pulse-type errors were successfully performed (Upadhyaya, et al., 1987a). In this section we discuss the results of application to a transient temperature data with simulated bias and noise-type sensor degradations. In addition to graphical displays of inconsistency index, log likelihood ratio and signal mean value, a tabular display of sensor status is given for all the signals simultaneously and updated at desired intervals.

Figure 2 shows the five redundant signals with bias, noise or both errors in various signals. The performance parameter set is given below:

- Missed and false alarm probabilities \( \alpha=0.01, \beta=0.01 \).
- Average mean time between false alarms for MSFRT is \( T=10,000 \) samples.
- Tolerance for the GCC, \( \eta_1=3^\circ F \).
- Tolerance for the bias SPRT, \( \nu_1=4^\circ F \).
- Normal and degraded signal variances, \( \sigma_0^2=1^\circ F^2, \sigma_1^2=4^\circ F^2 \).
Signal estimate, LLR, inconsistency index, and signal exclusion index are calculated at each sample. The signal estimate is shown in Fig. 3.

At about 10 sec. into the transient, errors were introduced into the signals. As an example of the estimation of performance signatures Figs. 4, 5 and 6 show the LLR for bias error, LLR for noise error and the inconsistency index for signal No. 4 respectively. The samples at which signal No. 4 is excluded are also shown in Fig. 6. Both types of degradations were detected in all the cases. In practice there are instances when detection of an error is possible but not the isolation. Further details of the GCC-SPRT and detailed discussion of the analysis are given in Glockler et al. (1987).

4. DIAGNOSTICS APPLIED TO A ROLLING MILL

In this section we want to present some results of applying multivariate autoregression (MAR) modeling to the diagnostics of a rotating machinery system. The details of the MAR modeling and estimation of spectral domain signatures are given in Glockler and Upadhaya (1987).

4.1 Multivariate Data-Driven Modeling

A set of stationary signals (fluctuating components) from a rotating machinery system may be modeled in the form

$$X(t) = \sum_{i=1}^{n} A_i X(t Li t) + V(t)$$

where $X(t) = [x_1(t), x_2(t), \ldots, x_m(t)]^T$ and $V(t)$ is a $m$-vector of driving noise sources, with finite covariance $E$. The purpose here is to determine the best parameter matrices $[A_i]$ such that Eq. (12) will characterize the system described by $X(t)$ as closely as possible.

The following spectral domain quantities are determined directly from the model by transformation to the frequency domain (Glockler and Upadhaya, 1987).

(a) Auto and cross power spectral density functions (APSD, CPSD).
(b) Ordinary and partial coherence functions (COH, PCOH).
(c) Ordinary and partial noise signal contribution ratios (NSCR, PNNSCR).
(d) Phase and transfer functions.

A systematic interpretation of these quantities will be used to detect and isolate sensor maloperation and process anomaly (mechanical components, controllers, etc.). It is important to note that this statistical analysis is applied to the fluctuating components of process signals.

4.2 Application to an Aluminum Rolling Mill

A very interesting application of MAR modeling to a process control system is discussed here. Several process signals from the Alcoa Hot Line rolling mill facility in Alcoa, Tennessee were recorded during more than fifty different runs. The process is not continuous, and each block of aluminum takes about 4 minutes to be rolled through the five-stand tandem mill. The following specific signal combination is modeled using a 40-th order MAR model.

(a) Exit sheet thickness.
(b) Stand five roll force.
(c) Stand four roll force.
(d) Interstand 4-5 sheet tension.

Figure 7 shows the APSDs of exit sheet gauge and stand-5 roll force fluctuations. Both of these APSDs have dominant spectral contributions at frequencies 3.4 Hz and 6.8 Hz. The corresponding phase and coherence are shown in Fig. 8. The coherence and partial signal contribution, PNNSCR, (Fig. 9) indicates that the primary driving source at the above frequencies is due to the changing work roll gap (very small) caused by the out-of-roundness in the work rollers. The fundamental rotating frequency of stand-5 work rolls is 1.7 Hz. The work roll gap fluctuations are reflected in the sheet exit gauge signal. The resulting vibration is transmitted to the roll force load cells. The analysis indicates that there are several perturbation sources in this system, and that the characteristic resonances in the signals are caused not by the sensor vibration, but by actual mechanically transmitted effects.

The coherence and phase indicate the interaction between two types of components in the signals-discrete frequency resonance effects and broadband process inherent random noise.
Sometimes these two effects may oppose each other, resulting in low coherence between a pair of signals (see Fig. 8). The linear phase behavior is due to the sheet propagation dynamics, with a time delay of $\tau=0.5$ sec from the location of the loadcell detector to the x-ray sheet gauge detector. This application strongly indicates that complex process dynamics can be represented by properly designed data-driven modeling of signal dynamic components.

5. CONCLUSIONS

A comprehensive approach to signal validation by processing of low-frequency and high-frequency components of process signals is described. The combination of GCC and modified SPRT is discussed and a few aspects of the algorithm are presented. The application of multivariate data-driven modeling is applied for the first time to the rotating machinery system of the Alcoa Hot Line facility. Even though this application is not one of sensor failure detection, the technique is equally applicable to sensor and process anomaly detection. The application indicates that it is possible to isolate the source of anomaly as being not the measurement system. This is a very important outcome of this research. This method may be applied to nuclear power plant components by proper identification of signals representing a subsystem.

The continuing work in this area includes the development of an expert system that will incorporate all the information generated by the algorithms and perform a low false-alarm failure diagnostics. For multichannel signal validation, a network of single channel GCC modules will be developed, and a logic will be devised to connect the outputs of multichannel modules. The microcomputer-based algorithms are suitable for in-plant applications and are developed as a portable system.

ACKNOWLEDGMENTS

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REFERENCES


Fig. 1. Schematic of the signal validation flow diagram using analytical and direct measurements, and combining the GCC and SPRT units.

Fig. 2. Five redundant temperature (deg F) signals with bias, noise, or both errors in various signals.

Fig. 3. The estimate of the process variable based on the five redundant signals.

Fig. 4. Log likelihood ratio of the sequential probability ratio test for bias error in signal No. 4.

Fig. 5. Log likelihood ratio of the sequential probability ratio test for noise error in signal No. 4.

Fig. 6. Inconsistency indices for signal No. 4. The samples at which signal No. 4 is excluded are represented by symbols on the graph.
Fig. 7. (a) Power spectral density function of the exit sheet thickness fluctuations (solid line) and that of its inherent noise component (dashed line) in a tandem rolling mill system.

Fig. 7. (b) Power spectral density function of the stand-5 roll force fluctuations (solid line) and that of its inherent noise component (dashed line).

Fig. 8. Coherence and phase functions between the exit sheet thickness fluctuations and the stand-5 roll force fluctuations. The coherence between the inherent noise components is zero; the phase is shown by the dashed line.

Fig. 9. Partial noise source contribution ratio functions (a) from the exit sheet thickness fluctuations to the stand-5 roll force fluctuations; (b) from the stand-5 roll force fluctuations to the exit sheet thickness fluctuations.
NEW APPROACHES IN SIGNAL PROCESSING TECHNIQUE FOR SODIUM BOILING NOISE DETECTION

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Abstract - The paper deals with the investigations carried out on signal processing techniques for developing a suitable pattern recognition method for sodium boiling noise detection in liquid metal fast breeder reactors. The investigations are particularly related with the suitable feature selection, threshold selection and evaluation of the quality of the signal processing techniques in terms of their capability in detecting sodium boiling with desired probabilities of spurious boiling and missing the boiling detection. The methods evolved have been tested on available experimental data.

1. INTRODUCTION

In liquid metal fast breeder reactors (LMFBRs), the detection of sodium boiling in its incipient stage is an important requirement from safety point of view. During sodium boiling, the generation, growth and collapse of sodium vapour bubbles affect the neutron noise of the reactor and lead to the emission of acoustic noise. Therefore, the detection of sodium boiling by monitoring the acoustic and neutron noise is considered to be a promising method. However, for reliable detection, it is necessary to distinguish the sodium boiling noise from the background noise in the reactor. Therefore, suitable signal processing techniques are to be evolved to detect the sodium boiling noise against the reactor background noise.

We carried out the investigations on the comparative assessment of signal processing and analysis techniques to identify or develop a new optimum method for a reliable on-line sodium boiling noise detection system. In this attempt, new features sensitive to boiling have been identified and new methods of threshold selection and evaluation of the quality of the signal processing techniques in terms of their capability in detecting sodium boiling with desired probabilities of spurious boiling and missing the boiling detection have been evolved. The study pertains to single as well as multiple features used in pattern recognition. The paper presents the details and results of these investigations. The present study has been conducted as part of the International Atomic Energy Agency (IAEA)'s Coordinated Research Programme (CRP) on Signal Processing Techniques for Sodium Boiling Noise Detection (Effiminko, 1983). The test data used in this study were made available by IAEA and are from experiments on KNS 1 loop (Hubber et al., 1976) of Kernforschungs zentrum Karlsruhe. These data were prepared at KFK (Rohrbachur and Aberle, 1984) by mixing boiling noise and background flow noise signals. The supplied data are in the form of 12-
files. Each file consists of three regions, namely, flow noise, initial boiling noise and intense boiling noise. In successive files, the signal to noise ratio (S/N) is reduced by 3 dB so that the first file data are with S/N=0 dB and the twelfth file data are with S/N = -17.5 dB. The duration of each file is 105 seconds and the data were recorded with a tape speed of 30 inch per second.

In Section 2 below, we discuss the methods of feature selection in probability density function (PDF) analysis, the method of selection of the boiling threshold, the target limits and the approach of achieving the target limits of the probabilities of spurious and missing the boiling noise detection and the results of the detection of onset of boiling and their comparison with the true values. The results of the extension of these procedures to power spectral density (PSD) and root mean square (RMS) value are also presented in Section 2. The above procedures have also been extended to pattern recognition methods where multiple features are considered simultaneously in the analysis. Section 3 provides the theoretical formulation and the results of sodium boiling noise detection on KNS data using this multiple feature pattern recognition technique. In Section 4, we give the conclusions.

2. SINGLE FEATURE SIGNAL PROCESSING

The analogue signals have been digitised using a 12 bit analogue to digital converter (ADC) with a minimum sampling time of 52 μs. To preserve the high frequency component of the sodium boiling noise, the tape was run at a reduced speed of 15/8 inch per second. The first and twelfth files of the analogue KNS-I data have only been considered for analysis. The background and boiling noise of the first file is used to learn their characteristics.

2.1. Probability Density Function Analysis

In order to calculate the Probability Density Function (PDF), the data have been sorted out into 64 equally spaced slots of pulse heights. Fig.1 depicts the PDF of background and background plus boiling noise and it can be observed that the former is more sharply peaked than the latter.

![Fig.1 PDF of the Signals (File-1)](image)

It is a feature commonly observed in the PDF analysis. However, a closer
scrutiny of the figure reveals that the areas under the PDF curve below slot number 27 and above the slot number 33 increased and the area in the slot range 28 to 32 decreased in the presence of boiling noise. The statistical variation in the areas marked A1, A2 and A3 is checked by calculating the mean and standard deviation values of the areas from large number of PDF estimates (See Table 1).

<table>
<thead>
<tr>
<th>Feature</th>
<th>Background noise</th>
<th>Boiling noise</th>
</tr>
</thead>
<tbody>
<tr>
<td></td>
<td>m1</td>
<td>σ1</td>
</tr>
<tr>
<td>A1</td>
<td>0.742</td>
<td>1.241</td>
</tr>
<tr>
<td>A2</td>
<td>1405.6</td>
<td>37.9</td>
</tr>
<tr>
<td>A3</td>
<td>188.8</td>
<td>40.8</td>
</tr>
</tbody>
</table>

m = mean value; σ1 = absolute standard deviation
σ1' = relative standard deviation = σ1/(m2 - m1)

The mean values of areas A1, A2 and A3 under PDF curves, show better discrimination for boiling noise. The values of the relative standard deviation as defined in the table, indicate that the distribution of feature estimates in boiling noise region is more scattered than in the background noise region. The similar calculations for twelfth file data (see Table 2) indicate that the feature estimates A1, A2 and A3 remain distinctively sensitive to the sodium boiling noise even in poor signal to noise ratio of -17.5 dB.

<table>
<thead>
<tr>
<th>Feature</th>
<th>Background noise</th>
<th>Boiling noise</th>
</tr>
</thead>
<tbody>
<tr>
<td></td>
<td>m1</td>
<td>σ1</td>
</tr>
<tr>
<td>A1</td>
<td>0.110</td>
<td>0.538</td>
</tr>
<tr>
<td>A2</td>
<td>1027.5</td>
<td>27.4</td>
</tr>
<tr>
<td>A3</td>
<td>433.10</td>
<td>37.30</td>
</tr>
</tbody>
</table>

However, the values of the relative standard deviations indicate that the spread in the distribution of feature estimates is more in this case and hence would result in larger overlap of the distributions of background and boiling noise.

2.1.1. Boiling Threshold Selection: The selection of boiling threshold for the new features A1, A2 and A3 is evaluated by calculating the PDF of these estimates in the background and boiling noise regions. The results could be as either of the three situations shown in Fig.2. Fixing up of the threshold in case (a) of the figure is obviously straightforward. However, for cases (b) and (c) when the distributions overlaps, whatever the threshold one chooses, it is not possible to avoid missing of the signals or getting spurious signals. Once a threshold is chosen any where between the mean values of the feature estimates in the background and sodium boiling noise regions, the fraction of the area of PDF curve of background noise above the threshold level will give the probability of spurious boiling detection, Ps and the fraction of the area of the PDF curve of boiling region below the threshold level would give the probability of missing, Pm, the boiling noise signal. To arrive at a most suitable threshold level, we employ the criterion that for a given threshold the product of the
probability of non-spurious boiling (1−Ps) and the probability of sodium boiling detection (1−Pm) is maximum. The values of the product (1−Ps) (1−Pm) for file 1 data and as a function of threshold is depicted in Fig. 3 which shows that as the threshold level is shifted from left to the right, the said product increases attains a peak for certain threshold and then starts decreasing. The values of the threshold levels, Ps and Pm corresponding to the fig. 3 and file 1 are listed in Table 3 and for file 12, in Table 4.

**Table 3**

<table>
<thead>
<tr>
<th>Feature</th>
<th>Threshold</th>
<th>Ps</th>
<th>Pm</th>
</tr>
</thead>
<tbody>
<tr>
<td>A₁</td>
<td>4.008</td>
<td>0.00426</td>
<td>0.12140</td>
</tr>
<tr>
<td>A₂</td>
<td>1281.9</td>
<td>0.00055</td>
<td>0.00200</td>
</tr>
<tr>
<td>A₃</td>
<td>281.7</td>
<td>0.01143</td>
<td>0.04845</td>
</tr>
</tbody>
</table>

**Table 4**

<table>
<thead>
<tr>
<th>Feature</th>
<th>Threshold</th>
<th>Ps</th>
<th>Pm</th>
</tr>
</thead>
<tbody>
<tr>
<td>A₁</td>
<td>0.777</td>
<td>0.108</td>
<td>0.392</td>
</tr>
<tr>
<td>A₂</td>
<td>1018.0</td>
<td>0.370</td>
<td>0.369</td>
</tr>
<tr>
<td>A₃</td>
<td>464.2</td>
<td>0.202</td>
<td>0.216</td>
</tr>
</tbody>
</table>
2.1.2. Target Limits on $P_s$ & $P_m$: Automatic trip systems for plant protection requires stringent targets for spurious trip and missing the boiling signal for trip. In U.K., an integrated spurious trip-rate of less than 0.1 per annum is demanded together with a probability of failing to demand of less than $10^{-3}$ (Ledwidge et al., 1986). Therefore, for the required maximum spurious trip rate of 0.1 per annum, a probability of spurious boiling, $P_s$, less than $1.9 \times 10^{-7}$ of tripping is needed for each 60 second decision time.

2.1.3. A New Method of Achieving the Target Limits: As can be seen from Tables 3 and 4, the values of $P_s$ and $P_m$ for the features $A_1$, $A_2$ and $A_3$ are much above the target limits. To reduce $P_s$ and $P_m$, a new approach is followed in taking the decision about the boiling when the signal crosses the present threshold level. It is based on the fact that if $P_s$ is the probability of spurious boiling detection for signal estimate crossing the threshold level and out of no feature estimates evaluated at a time, features estimate show boiling, than the probability of spurious boiling detection would be given by the following binomial distribution, i.e.

$$P'_s(n, n_0) = \frac{n_0!}{n!(n_0-n)!} P_s^n (1-P_s)^{n_0-n}$$

For out of no estimates, if no or more estimates indicate boiling then the probability of spurious boiling would be

$$P_s(n_0, n_0) = \sum_{n=n_0}^{n_0} \frac{n_0!}{n!(n_0-n)!} P_s^n (1-P_s)^{n_0-n}$$

Similarly if $P_m$ is the probability of missing the boiling for a single
record showing no-boiling and out of No estimates evaluated at a time, n estimates indicate no-boiling, then the probability of missing the boiling would be

$$P_m(\frac{1}{n}, N_0) = \frac{N_0!}{n! (N_0-n)!} P_m^\frac{n}{(1-P_m)^{N_0-n}}$$

(3)

From out of No estimate, if no or less estimate indicate no-boiling, then the probability of missing the boiling would be

$$P_m(\frac{1}{n}, N_0) = \sum_{n=0}^{N_0} \frac{N_0!}{n! (N_0-n)!} P_m^\frac{n}{(1-P_m)^{N_0-n}}$$

(4)

Therefore, to reduce the probability of spurious or missing the boiling detection, decision for boiling detection should be taken based on not a single estimate but on No-estimates evaluated at a time and by checking whether no or more estimates show the boiling for the present threshold value. This we will refer to as no/No criterion for boiling detection. The effectiveness of the procedure becomes more clear in the following.

2.1.4. Detection of Onset of Boiling: The probability density function analysis through the feature estimates A_1, A_2 and A_3 has been applied to detect the point of onset of boiling for the test data of file 1 and file 12. The results are given in Table 5 along with the values of no/No to achieve the target probabilities and the true values of the onset of boiling.

<table>
<thead>
<tr>
<th>Feature</th>
<th>File 1</th>
<th>File 12</th>
</tr>
</thead>
<tbody>
<tr>
<td></td>
<td>Predicted</td>
<td>True</td>
</tr>
<tr>
<td>A_1</td>
<td>5/10 29.0 s</td>
<td>27.0</td>
</tr>
<tr>
<td>A_2</td>
<td>3/4 32.7 s</td>
<td>27.0</td>
</tr>
<tr>
<td>A_3</td>
<td>6/9 34.4 s</td>
<td>27.0</td>
</tr>
</tbody>
</table>

*target values for spurious and missing the boiling detection are 0.32 x 10^-8 and 10^-3 respectively.

The threshold level, Ps and Pm values used in the calculations are same as given in Table 3 and 4. The difference between the predicted and true values may be due to the fact that the noise characteristics for initial and fully developed boiling stages may differ and only fully developed boiling was used for fixing the threshold. Feature estimate A_1 is found more sensitive to the boiling noise than the other feature estimates.

2.2. Power Spectral Density Analysis

The above methodology is applied to power spectral density (PSD) also. The PSD calculated in the background and boiling regions is shown in Fig.4 which indicates that 35 to 140 kHz region is sensitive to the sodium boiling noise. So the total PSD in this frequency window was taken as a discriminating feature boiling and background region. The threshold level, Ps and Pm were calculated as in the case of A_1, A_2 and A_3 features. The method applied to get the time of onset of boiling for file 1 and file 12 predicted boiling at 27.55 sec. for first file and 26.4 second for twelfth file while the corresponding true values are 27.0 and 27.8 seconds respectively. One more noteworthy feature of this feature is that once
onset of boiling is predicted, all subsequent decisions indicated boiling.

2.3. Root Mean Square (RMS) Analysis

The above methodology was applied to RMS also. Two sets of RMS analysis were done. One is with the data as such and the second with the squared data. The results for the prediction of onset of boiling by this method for file 1 and 12 are given in Table 6.

Table 6
Prediction of Onset of Boiling Time (s): Different Methods*

<table>
<thead>
<tr>
<th>Method Tested</th>
<th>File 1</th>
<th></th>
<th>File 12</th>
<th></th>
</tr>
</thead>
<tbody>
<tr>
<td></td>
<td>Predicted</td>
<td>True Values</td>
<td>Predicted</td>
<td>True Values</td>
</tr>
<tr>
<td>PDF A_1</td>
<td>29.0</td>
<td>27</td>
<td>31.4</td>
<td>27.8</td>
</tr>
<tr>
<td>PDF A_2</td>
<td>32.7</td>
<td>27</td>
<td>42.9</td>
<td>27.8</td>
</tr>
<tr>
<td>PDF A_3</td>
<td>34.4</td>
<td>27</td>
<td>34.3</td>
<td>27.8</td>
</tr>
<tr>
<td>RMSNormalData</td>
<td>31.9</td>
<td>27</td>
<td>34.1</td>
<td>27.8</td>
</tr>
<tr>
<td>RMSSquaredData</td>
<td>31.1</td>
<td>27</td>
<td>35.9</td>
<td>27.8</td>
</tr>
<tr>
<td>PSD</td>
<td>27.6</td>
<td>27</td>
<td>26.4</td>
<td>27.8</td>
</tr>
</tbody>
</table>

* Target limits for the probability of spurious boiling detection .32 x 10^{-8} and for missing the boiling detection 1.0 x 10^{-3}.

For the sake of comparison, the results of other methods are also given in this table. It can be seen that for all the features considered, the PSD and A_1 feature for predict the onset of boiling very near the real values.

3. A NEW MULTIVARIATE PATTERN RECOGNITION TECHNIQUE

The proceedings of Section 2, have been extended to pattern recognition methods where multiple features are considered simultaneously in the analysis.
3.1. Theoretical Formulations

Let us consider \( N \)-features that can discriminate boiling and non-boiling signals and let for the \( i \)th feature \( F_i \), the probabilities of spurious and missing the boiling signal is \( P_{si} \) and \( P_{mi} \) respectively. If all the features are completely independent, then the probability of getting the spurious or missing the signals would be,

\[
\begin{align*}
P_s &= \prod_i P_{si} \quad \text{and} \quad P_m = \prod_i P_{mi}.
\end{align*}
\]  

(5)

However, in general all the features may not be independent and may have some correlations among them. In such a case, it would not be so easy to get \( P_s \) and \( P_m \) as by Eq. (5).

In the case of single feature, the distributions of the feature in background and boiling regions lie on a real line and threshold is a point on this real line. In the case of multi-feature analysis, the probability distribution would be in \( N \)-dimensional space. If the features are all independent, then the distribution of features is given by,

\[
P(\vec{A}) = \prod_i \frac{1}{(2\pi)^{\frac{N}{2}} \sigma_i} e^{-\frac{1}{2} \left( \frac{A - \mu_i}{\sigma_i} \right)^2}
\]

where \( A \) is the \( N \)-dimensional vector, \( \mu_i \) is the \( i \)th component of the estimate and \( \sigma_i^2 \) are the \( i \)th component mean and standard deviation respectively. Now the threshold would be a surface in a \( N \)-dimensional space. If an estimate falls outside this surface, it would be declared as boiling otherwise non-boiling. For correlated signals, the distribution of an estimate is given by multivariate Gaussian function,

\[
P(\vec{A}) = \frac{1}{(2\pi)^{\frac{N^2}{2}} |\mathbf{C}|^{\frac{1}{2}}} e^{-\frac{1}{2} (\vec{A} - \vec{\mu})^T \mathbf{C}^{-1} (\vec{A} - \vec{\mu})}
\]

(7)

where \( \vec{\mu} \) is the mean of the vector estimate, \( \mathbf{C} \) is the covariance matrix and \( |\mathbf{C}| \) is its determinant. \( (\vec{A} - \vec{\mu})^T \) is the transpose of the vector \( \vec{A} - \vec{\mu} \). The expressions of relative standard deviations in this case would be,

\[
\begin{align*}
\mathbf{C}_N' &= \left[ (\vec{\mu}_1 - \vec{\mu}_2)^T \mathbf{C}_1^{-1} (\vec{\mu}_1 - \vec{\mu}_2) \right]^{-\frac{1}{2}}
\end{align*}
\]

(8)

\[
\begin{align*}
\mathbf{C}_B' &= \left[ (\vec{\mu}_1 - \vec{\mu}_2)^T \mathbf{C}_2^{-1} (\vec{\mu}_1 - \vec{\mu}_2) \right]^{-\frac{1}{2}}
\end{align*}
\]

where \( \mathbf{C}_N' \) and \( \mathbf{C}_B' \) have the same significance as \( \sigma_i' \) and \( \sigma_2' \) in Table 1. \( 1/\mathbf{C}_N' \) and \( 1/\mathbf{C}_B' \) are nothing but the Mahalanobis distances and for uncorrelated features, the Mahalanobis distance is nothing but the cartesian distance between \( \vec{\mu}_1 \) and \( \vec{\mu}_2 \) where the axis have been normalised with corresponding standard deviations.

3.1.1. Calculations of \( P_s \) and \( P_m \): Let \( \vec{\mu}_1 \) and \( \mathbf{C}_1 \) be the mean vector and covariance matrix of \( N \) features in background region and \( \vec{\mu} \) and \( \mathbf{C}_2 \) the corresponding parameters in boiling region. If \( S \) be the surface in \( N \)-dimensional space separating boiling and non-boiling regions, then,

\[
P_s = 1 - \frac{1}{(2\pi)^{\frac{N^2}{2}} |\mathbf{C}_1|^{\frac{1}{2}}} \int_{\vec{A}} e^{-\frac{1}{2} (\vec{A} - \vec{\mu})^T \mathbf{C}_1^{-1} (\vec{A} - \vec{\mu})} d\vec{A}
\]

(9)
\[ P_n = 1 - \frac{1}{(2\pi)^{N/2} |C|^2} \int_{m_2}^{S} e^{-\frac{1}{2} (\vec{n} - \vec{m}_2)^T C^{-1} (\vec{n} - \vec{m}_2)} d\vec{n} \]  

(10)

3.1.2. Threshold Fixing: Let us draw a line from \( \vec{m}_1 \) to \( \vec{m}_2 \) in the \( N \)-dimensional space and \( T \) be an arbitrary point on that. An \( N \)-dimensional sphere is draw with \( \vec{m}_1 \) as the centre and Mahalanobis distance between \( \vec{m}_2 \) and \( T \) as its radius, the covariance matrix being corresponding to the non-boiling region. A similar sphere is also drawn with \( \vec{m}_2 \) as centre and with the covariance matrix of the boiling region. Now if any feature vector falls outside the first sphere then it is taken as an indication of boiling. In that case, the probability of spurious counts would be,

\[ P_s = \int_{T_1}^{\infty} dT (2\pi)^{-N/2} |C_1|^{-1/2} e^{-\frac{T}{2}} \]

where \( T \) is Mahalanobis distance and

\[ C = (T - \vec{m}_1)^T C_1^{-1} (T - \vec{m}_1) \]

(12)

In a similar way, if a feature estimate falls in the second sphere it is declared as boiling. The probability of missing the boiling signal would be,

\[ P_m = \int_{T_2}^{\infty} \frac{1}{|C_2|^{-1/2}} e^{-\frac{T}{2}} dT \]

where

\[ C_2 = (T - \vec{m}_2)^T C_2^{-1} (T - \vec{m}_2) \]

(13)

(14)

\( P_s \) and \( P_m \) are calculated as a function of \( T \) and the value of threshold for boiling is taken when \((1-P_s)(1-P_m)\) is maximum.

3.1.3. Application to Boiling Detection: Application of the method is tested on KNS-I data again. The areas under the PDF curves \( A_1, A_2 \) and \( A_3 \), the second, third and fourth moments are the features considered for the analysis. Due to the computing time limitations, the PSD is not considered as a feature in time analysis. The calculations of correlation coefficients of the features in the background and boiling regions revealed that the second and fourth moments and second moments and \( A_3 \) are highly correlated and correlation is more pronounced in the boiling region. So one of the features become redander and the features are selected such that the correlation among them is minimum. Thus instead of taking second, third and fourth moments, standard deviation, skewness and the kurtosis are taken as the features. The procedure of fixing the threshold mentioned above is followed and for file 1 data we found a region for threshold which satisfies the criteria that \( P_s < 10^{-8} \) and \( P_m < 10^{-3} \). This means that the distribution of points in six dimensional space is well separated for background and boiling regions. For the data of file 12, we do not find such a threshold point which satisfy the target \( P_s \) and \( P_m \). So in order to achieve the target values we applied no/No criteria and for 15/16 we got the target values for \( P_s \) and \( P_m \). For file 1, the predicted time for onset of boiling is 27.4 sec. and for file 12, with 15/16 criteria, the onset of boiling time predicted is
4. CONCLUSIONS

One can draw the following conclusions from this study.

(i) Segmented areas under the PDF curves are sensitive to the sodium boiling noise and hence can be considered as good features for sodium boiling noise detection.

(ii) Maximisation of the probabilities of boiling and non-boiling detection for given thresholds is suitable for fixing the thresholds for boiling noise detection.

(iii) The use of binomial distribution as suggested for achieving the low probabilities of spurious/missing the boiling detection and applied to the prediction of onset of boiling in KNS-I data can be used to achieve quite low target probabilities.

(iv) Relative standard deviation defined in Table 1 is useful for assessing the relative overlap of the PDF of the features for background and boiling noise.

(v) PSD and $A_1$ features are found to be quite sensitive to predict the onset and subsequent boiling.

(vi) The use of multiple features at a time adds reliability in the prediction.

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SIGNAL PROCESSING TECHNIQUES FOR THE ACOUSTIC DETECTION OF BOILING IN LMFBRS
PRELIMINARY RESULTS OF AN IAEA COLLABORATIVE RESEARCH PROGRAMME

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ABSTRACT

A specialist meeting organised by the International Working Group on Fast Reactor (IWGFR) of the International Atomic Energy Agency (IAEA) was held at Chester in the United Kingdom in 1981 to discuss techniques for the detection of acoustic noise from boiling. This meeting recommended that a benchmark test should be carried out to evaluate and compare signal processing methods for use in the detection of the acoustic noise produced by boiling sodium. In response to this recommendation the IAEA set up a collaborative research programme to examine and compare the processing techniques used in the laboratories of member countries. Eight laboratories in six countries have taken part in the programme which will be completed in 1988. This paper summarises the results obtained so far.

INTRODUCTION

In 1981 a specialist meeting, initiated by the International Working Group on Fast Reactors (IWGFR) of the International Atomic Energy Authority (IAEA), was held to review methods of signal processing for use in the detection of the acoustic noise produced by boiling in sodium. (1). This meeting concluded that acoustic detection showed considerable promise of making a useful contribution to reactor safety and recommended that Benchmark Exercise be carried out to compare the signal processing methods under development in the various participating countries.

As a result the IAEA called a meeting of consultant specialists in 1983. This meeting drew up proposals for the first stage of the test which used data obtained from the ENS series of sodium boiling experiments carried out at Kernforschungszentrum Karlsruhe in the Federal Republic of Germany. These data were provided by KfK Karlsruhe and NRL, UKAEA, Risley, UK who had collaborated in the acoustic measurements on these experiments.

The data were distributed to participants in 1984 and a meeting was held in 1985 to consider the first results. At this meeting a further set of test data was offered by the Central Institute for Nuclear Research, Rossendorf in the German Democratic Republic. This set of data was from a series of boiling experiments carried out in the BOR60 reactor in the USSR. The participating Laboratories are listed in the Table together with the acronym which is used to identify them in the text.
TABLE

List of Participating Laboratories

<table>
<thead>
<tr>
<th>Acronym</th>
<th>Laboratory</th>
</tr>
</thead>
<tbody>
<tr>
<td>IGCAR</td>
<td>Indira Chandi Centre for Atomic Research, India</td>
</tr>
<tr>
<td>DDAE</td>
<td>Darling Downs Institute of Advanced Education, Australia</td>
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<tr>
<td>CINRR</td>
<td>Central Institute for Nuclear Research, Rossendorf, GDR</td>
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<tr>
<td>KfK</td>
<td>Kernforschungszentrum Karlsruhe, FRG</td>
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<td>IA</td>
<td>Interatom, Bensberg, FRG</td>
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<td>BNL</td>
<td>Berkeley Nuclear Laboratories, CEGB, UK</td>
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<td>NRL</td>
<td>Northern Research Laboratories, UKAEA, Risley, UK</td>
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<td>JAERI</td>
<td>Japan Atomic Energy Research Institute, Tokyo, Japan</td>
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The analysis of these signals has been carried out independently in the participating laboratories but, as would be expected, there is a broad area of agreement as to the approach. Therefore the work is described in this paper without attribution to particular laboratories. Individual laboratories are identified only in connection with aspects which are unique to them or have been pursued particularly deeply there.

This paper describes progress up to December 1986 at which point each laboratory had analysed the test data though it is only possible to give a brief illustration of the work carried out rather than a full critical appraisal. The programme will continue into 1988 to compare the methods quantitatively and to produce a recommendation for an optimum processing system for boiling noise detection.

DESCRIPTION OF TEST DATA

Distribution Medium

The sets of test data were distributed in the form of analogue records on magnetic tape. The acoustic signals were recorded in the direct mode at a tape speed of 30 ips to IRIG Intermediate Band standards. In addition to the acoustic data two channels were used for time codes. These channels were recorded in the frequency modulation mode to IRIG Wideband 1 standard. The time codes enabled all the analysis teams to have a common time reference by which to define features in the data.

KNS Experiment

These data sets were taken from records made during tests with the first KNS test section in which boiling was caused to occur behind a central planar blockage in a simulated SNR Reactor fuel subassembly. Experiments with this test section have been described by Huber et al. (2).

In the experiment the rate of sodium flow through the test section is initially high so that there is no boiling. The flow is then reduced by steps to produce first incipient local boiling and then progressively till there is fully developed boiling in the region behind the blockage. The acoustic signals were obtained from an accelerometer mounted on a metal rod attached to the test section which served as a waveguide.

The benchmark test consisted of twelve tape records. The first was the original record obtained from the accelerometer referred to above. The succeeding eleven records contained the same signal but with successively increasing levels of background noise. These records were produced artificially by adding background noise taken from the output of the same transducer in high flow non-boiling conditions, to the signal to give successive reductions of about 2db in the signal to noise ratio for each file so that the signal to noise ratio in the last record was some 17.5 db below that of the first. The time at which the boiling commenced was varied artificially relative to the start of each of these records.

The participants were asked to use their analysis techniques to determine the times of the onset of boiling and the major increase in boiling noise level within these records.

BOR60 Experiment

The data for this experiment was supplied by the Central Institute for Nuclear Research, Rossendorf, DDR. In the experiments boiling was produced within a cluster of tungsten rods which were placed within the core area of the reactor. Heat was generated in the tungsten principally by gamma ray absorption. The experiment has been described by Afanas’ev et al. (3).
Data were provided from two test rig configurations each having the same pin cluster in which boiling is produced but different arrangements of transducers. In the first the acoustic detectors were mounted at about core top height but within the experimental test rig. In the second rig the acoustic detectors were quite separate from the boiler rig providing a much more realistic simulation of the conditions in which an acoustic detection system would be used.

The analysis procedures adopted by all the laboratories can be considered in three stages. First it is necessary to determine the principal characteristics of the boiling noise signal and of the background noise in order to select those which offer greatest promise of being capable of development into a sensitive and reliable boiling noise detection system. The second stage is the development of signal processing algorithms to maximise the reliability and sensitivity of the discrimination achieved through the features identified in the first stage. Finally it was necessary to select criteria for making the decision as to whether boiling existed. This was derived in the form of a probability and the efficiency of an analysis technique is assessed from a comparison of the probability of a false indication of boiling, which might lead to an expensive spurious trip in a reactor, and the probability of failing to detect boiling.

These three phases of work existed in the programmes of all the participating laboratories though the emphasis given to each stage varied considerably. The main features of the work carried out under these three headings are described in the following sections.

**Determination of Signal Characteristics**

There was a high level of agreement among the participating laboratories as to what were the principal features which distinguished the boiling and background signals. These can be deduced as follows. Figure 1 shows samples of the boiling signal taken from the KNS and the BOR60 data.

![Fig. 1 Comparison of boiling signals from KNS and BOR60.](image)

It is clear that the sharp high amplitude pulses are an important and common feature of the two sets of data. Such pulses have a power spectral density, PSD, with strong high frequency components which, in principle, should enable the boiling signal to be recognised readily against low frequency background noise. In low level incipient boiling, however, the pulses are so far apart that their contribution is only a small part of the total noise. This can be seen in Fig. 2 which shows the PSD of noise alone and of the total signal when boiling is present. The boiling causes a detectable increase in the PSD at frequencies above about 10KHz but only by a relatively small margin.

In the case of low level boiling with widely spaced pulses amplitude discrimination techniques, such as pulse counting, offer greater sensitivity. This is illustrated in Fig. 3 which shows the variation of count rate of pulses exceeding a discrimination threshold as boiling develops in the KNS experiment. This simple application of amplitude discrimination is less sensitive when the normal background also contains impulsive noise signals as was the case in the data from the BOR60 experiment as shown in Fig. 4. Impulses in the background can be due to acoustic signals from cavitation or mechanical rattling or from electrical interference.
Fig. 2 Rig 1: Comparison of spectra for background and boiling for File 3.

Fig. 3 Comparison of RMS signal amplitude and pulse count rate for File 2 S/N 0dB.

Fig. 4 Comparison of boiling and non boiling signals from BOR60.
The characteristics which distinguish the boiling signal therefore are the presence of sharp high amplitude pulses in the time domain which give rise to strong high frequencies in the psd curve. Neither of these features however, used on its own, is sufficient to give a reliable and sensitive detector of boiling. The participating teams all found it necessary to use more complex processing techniques to improve sensitivity and reliability. This work is described in the following section.

Signal Processing Techniques

The first analysis technique investigated by most groups was the use of the rms value of the signal in a particular frequency band averaged over a suitable time interval. The second basic technique, pulse counting above a discriminator level, fixed at a level corresponding to say 6 times the rms value of the pre boiling background signal for the record, successfully detected boiling in each of the sets of KNS data.

Though both the above basic techniques detected boiling, neither seemed likely to give the best sensitivity for the detection of incipient boiling which maintaining the stringent requirement for a very low rate of spurious trips. Accordingly the participating laboratories explored the use of a number of more complex processing techniques.

One technique which enhances the impulsive nature of the boiling signal is squaring the signal. An extension of this method where repeated squaring is combined with threshold discrimination and high pass filtering is being investigated (JARRT). This system is shown in block diagram form in Fig. 5 and was used successfully to detect the high level of boiling in the BOR60 data.

![Block diagram of signal processing method](image)

**Fig. 5** Schematic diagram of the signal processing method

A number of laboratories examined the use of the higher order derivatives of the probability density function, skew and kurtosis and found the latter to be a sensitive indicator of the presence of impulsive noise in the signal. Closely related to kurtosis is the K function developed by CENRR described as a measure of the roughness of the signal waveform and defined as the ratio of the variance of the signal to the square of the mean value of the rms of the signal. Figure 6 shows how this parameter detects the onset of boiling in one of the KNS runs, the raw signal and the count rate are also shown for comparison.

DDIAE investigated the use of high order differences to detect the presence of boiling. The discriminant is defined by the equation:

\[
Z_r^2(j) = \frac{1}{N} \sum_{k=0}^{N-1} [Z_{r-1}(k-j) - Z_{r-1}(j-k-1)]^2
\]

in which \(z(j)\) is the jth element in the time series of values representing the signal.
This group also investigated a conditional expectation approach in which the mean square prediction error is given by:

$$\varepsilon^2(j) = \frac{1}{N} \sum_{k=0}^{N-1} [Z(j-k) - E(Z(j-k)|Z(j-k-1), \ldots, Z(j-k-n))]^2$$

where $E(.)$ is the expectation estimated using data in the non-boiling "learning" phase of the process.

Figure 7 shows a comparison between the simple mean square estimate, the expectation error and the 6th order difference.

To improve the signal to noise ratio by taking advantage of the impulsive nature of the boiling signal NRL used transient capture facilities to collect records of the signal which contained pulses and ignored the non-impulsive part of the background. These records contained boiling pulses and spurious pulses from the background. Pattern Recognition Methods, which were extensively studied at IGCAR and KFX as well as NRL, were used to separate these. In this technique a large number of features of the boiling signal and of the background noise are measured and algorithm such as the Principal Components algorithm is used to select which features or combination of features gives the best discrimination between the signal and the background. Features which were used in this process include bands in the power spectrum and the slope of the PSD curve, also features of the probability distribution such as skew and kurtosis. A typical result from the BOR60 data is shown in Fig. 8. In this case the features used are the power in the 50 to 62.5KHz band and the ratio of the powers in the 75 to 100KHz band and 50 to 75KHz band.
Another analysis method was the Adaptive Learning Techniques applied at BNL and KfK. The procedure used at BNL in creating an Adaptive Learning Network (ALN) consist of considering all possible combinations of two of the signal parameters into the first stage of a second order polynomial which is fitted to a target value viz:

$$x_{kl} = a_0 + a_1 x_{i1} + a_2 x_{j1} + a_3 x_{i1}^2 + a_4 x_{j1} x_{i1} + a_5 x_{j1}^2$$

(2)

where

- $x_{kl}$ is the building block output
- $x_{i1}$ is the $i$th feature
- $x_{j1}$ is the $j$th feature

Network training is achieved by estimating coefficients $a$ in (2) from the training set of data to minimise the error function given by:

$$(error)^2 = \sum_{l=1}^{L} (x_{kl} - t_l)^2$$

where $L$ is the number of samples in the training set. Figure 9 shows a summary of the detection failure and spurious trip probabilities estimated for this method. The analysis was performed using the amplitude characteristics of the signals eg. rms, skewness and kurtosis as input parameters to the network.
The data supplied from BOR60 included a signal from an ionisation chamber outside the core. The ionisation chamber signal is plotted in Fig. 10 along with the rms acoustic signal from the pressure transducer D1. The correlation between the two signals is clear. Such correlation in the behaviour of two independent parameters is strong confirmation of the presence of boiling and would provide a most valuable method of confirming a marginal indication of boiling from an acoustic channel. This encouraging result has however to be qualified by noting that the tungsten rods used to produce heat from gamma ray absorption in BOR60 would give a low heat flux with correspondingly low temperature gradients. This would result in larger vapour bubbles than might occur in a normal fuel channel in fault conditions, at least at the incipient boiling stage, so that the ionisation signal is probably large than would occur in that case.

Fig. 10 Comparison of signal from ionisation chamber and rms acoustic signal from microphone D1 for non-boiling and boiling conditions in File 9 (Fig.2).

Decision Criteria

The effectiveness of a boiling detection instrument is a combination of its sensitivity to detect boiling at low levels and its robustness against spurious trips generated by components of the background noise. In a working power station the cost of an unnecessary shutdown is substantial and the probability of one must be reduced to a very low level. This is sometimes specified as not more than one per year which means less than one in ten years for any one instrument.
DDIAE proposed that the quality of the discrimination between boiling and background achieved by a technique could be evaluated in terms of the probabilities of a wrong diagnosis that boiling is present and a failure to detect the presence of boiling. If the probability density function of the detection parameter is plotted for both the signal without boiling and the signal when boiling is present, then the probability of a wrong diagnosis can be calculated once the discriminating value of the parameter has been selected. If the discriminator is set at the point of intersection of the two distributions, the point at which the balance of probability changes from non-boiling to boiling the result can be an unacceptably high probability of a spurious trip $P_s$ defined by the area of the distribution for the background signals which lies to the right of the discrimination level. A typical result of this process for the BOR60 data yielded $P_s = 0.049$ and $P_m = 0.018$. These values can be improved by raising the discrimination level to achieve an acceptable probability of spurious trip at the expense of increasing the probability of failing to detect boiling $P_m$. IGCAR showed that by basing the decision as to whether boiling exists on a number of samples rather than just one the binomial expansion can be used to calculate the probability of failing to detect boiling and that by taking sufficient samples, that is increasing the decision time, the probability of failure can be reduced to as low a level as is required.

![Graph showing probability curves for feature No. 5 (power in 58-62.5 kHz band) showing boiling detection threshold.](image)

A similar approach is shown in Fig. 11, from NRL using the pulse sampling method, where for the BOR60 data the discrimination level has been set at six standard deviations of the background distribution thereby achieving the desired low rate of spurious trip. This is achieved however at the expense of a probability of failing to detect boiling in one out of eight samples when boiling exists. Since the rate of occurrence of boiling pulses is at least five per second this would result in a delay of only a fraction of a second in recognising boiling.

**DISCUSSION**

The specific task set for the laboratories in the bench mark test specification was to detect the onset of boiling and a rise in boiling power to a higher level. It should be noted that this higher level of boiling was still a stable regime which could safely be allowed to persist for much longer than the time required for detection. All techniques successfully detected the higher level of boiling even in the KNS case where noise had been added to reduce the signal to noise ratio by 17.5 db. The immediate conclusion from this is that the test was too easy and failed to show up the strengths and weaknesses of the techniques under examination. This is true and is one reason why in the later stages of the common research programme emphasis is being placed on the quality of the detection. There is however another message which is that boiling produces a good acoustic signal which should be detectable by relatively simple and reliable techniques. Similarly in the case of the BOR60 data the developed boiling produced a signal which could be detected relatively easily above the normal reactor background noise.

The detection of the early stages of incipient boiling also appears possible and several techniques gave results in good agreement with each other in this area. In the limit however the detection of the onset of incipient boiling reduces to an attempt to detect the pulse from the collapse of the first bubble. Such a target is neither realistic nor achievable in an operating system, though it is a nice exercise for a rig experiment or benchmark test. The detection of a single pulse would not achieve the required immunity from spurious trips, unless of course the
pulse was very large compared to the background noise. In the benchmark test some very sensitive systems emerged to detect the onset of incipient boiling but the work necessary to establish the quality of this detection has still to be completed.

The work done so far has shown that if the maximum sensitivity for the detection of incipient boiling is required sophisticated data acquisition hardware and computer data processing of fairly complex algorithms will be required. At the present time such equipment has not been used in safety systems. It is likely that technology for computer processing of this type in safety systems will become available, till then acoustic boiling noise detection may have to function at two levels, a trip function based on simple techniques to detect developed boiling and a diagnostic system producing a warning of incipient boiling to the operator. Such a system might also warn the operator of any correlation between the boiling signal and any other relevant parameter such as the neutron flux, reactivity, temperature, or temperature noise signals. Again we can look forward to this type of inter parameter comparison being carried out by computer in the future probably using an expert system.

The work carried out under the research programme has gone beyond the original idea of a benchmark test of existing techniques. The useful exchange of ideas which is taking place is likely to result in a final proposal for an optimum system which will embody contributions from all of the laboratories taking part.

CONCLUSIONS

The IAEA Coordinated Research Programme (CRP) under which this work has been carried out is not yet complete but the following provisional conclusions can be drawn.

1. The acoustic signal from boiling in sodium is highly impulsive and detection methods will make use of this feature either by detecting the impulses in the time domain or by detecting the resulting high frequency signals in the frequency domain.

2. While general conclusions cannot be drawn from the small amount of data studied in the test the results support the belief that acoustic techniques will be able to detect boiling at an acceptable level in an LMFBR.

3. Relatively simple processing is likely to be sufficient to provide trip protection but more sophisticated computerised data processing will be required to achieve maximum sensitivity and the earliest warning of an incipient fault condition.

4. The CRP is providing facilities for a useful cross fertilisation of ideas and should lead in 1988 to the proposal of an optimum processing technique for boiling noise detection in LMFBR's which will aim to meet the requirements of safety and the operational need for the earliest possible warning of a fault condition.

REFERENCES


FOURIER
TRANSFORMATION AND
PATTERN RECOGNITION

Session chairman: K. Behringer (Switzerland)
SUMMARY OF SESSION

This session, entitled 'Fourier Transform and Pattern Recognition', is concerned more with basic problems in noise analysis methods than with new applications. It contains five papers.

The paper of Pineyro and Behringer deals with a problem in the field of noise data qualification. A new method based on Fourier transform techniques is presented which allows one, within certain limitations, to distinguish between sinusoidal components and narrowband random noise contributions in otherwise random noise data. For this analysis a special 4th-order spectral function has been introduced. The paper gives the theoretical background of the analysis procedure, proposes a validation criterion for the type identification of the peaks in the PSD, and shows results from experimental examples. The method is believed to be suitable for semiautomatic routine applications.

Reddy and Murthy propose a new method for the on-line detection of a malfunction. The method uses noise data segmentation and requires the selection of a statistical feature variable which has to show amplitude distributions well separated under normal conditions and in the presence of a specific malfunction, so that a threshold can be defined. By summing instantaneous feature values relative to the threshold from preceding segments, a random walk is obtained which runs either to a left-hand boundary or to a right-hand boundary, indicating either the normal or the anomalous state, respectively. After the walker has crossed one of the boundaries it is reset and restarted. The method and the theoretical considerations given to the problem of spurious decisions have been tested on sodium boiling noise.

The aim of the paper of Dailey and Albrecht is to model the vibration of the core internals in an operating PWR, as sensed by the fluctuations of the core and ex-core neutron detector signals, and to develop a set of parameters for establishing a discriminant which allows monitoring of the state of vibrating internals. In the first part of the paper the modelling procedure is described. It can be considered as a continuation of work published at SHORN-IV. The model is a linear combination of 4 types of independent noise sources: lateral motions of the fuel assemblies, lateral motion of the core support barrel, background noise, and spatially localized assembly-specific information. In the second part of the paper a state vector is constructed. A statistical (frequency dependent) discriminant is proposed, which indicates the normalized squared deviation of the presently observed state from a baseline state. The methodology provides a means of establishing a baseline state, evaluating the deviation of the present state from the established baseline, and suggests a way of setting a threshold.

The appearance of an anomaly may be associated with a nonlinear random process which effects non-Gaussian contributions to the noise signal. For the characterization of non-Gaussian noise, higher-order correlation functions or their corresponding Fourier transforms, like the triple correlation function or the bispectrum, respectively, must be considered. The paper of Konno is concerned with modelling the category of hump and burst phenomena appearing in randomly excited nonlinear mechanical vibration systems. It has the aim of system identification by using bispectral techniques in addition to the amplitude distribution function. The given Langevin equation is based on a one-dimensional vibrational model and contains, as special cases, the equations which are shown to describe either the hump phenomena or the burst phenomena.
The paper of Avila and Oliveira refers to the analysis of core barrel vibrations in PWRs. The method of "identifying functions" is applied to ex-core neutron noise data from the Borssele reactor (SMORN-III benchmark data) and from 3 French PWRs. The method of "identifying functions" has recently been proposed by the authors and consists of the definition of 4 pairs of functions of frequency which are based on the phase relationships of the CFSDs between 4 ex-core neutron detectors located in different quadrants of the same plane. Each pair of these functions enhances the appearance of different noise sources. In particular, the application of this method to the Borssele PWR has resulted in the observation of shell mode core barrel vibrations which have not been reported earlier.
DISPLACED SPECTRA TECHNIQUES AS A TOOL FOR PEAK IDENTIFICATION IN PSD-ANALYSIS

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Abstract – Sharp peaks in the power spectral density function can be due to periodic components in the noise signal or due to narrowband random contributions. A novel method based on Fourier transform techniques is presented which allows under certain limitations to identify the peak type.

Keywords – Noise analysis, data qualification, power spectral density, peak identification, periodic components, narrowband random data.

1. INTRODUCTION

In the estimation procedure of the power spectral density (PSD) function by the finite Fourier transform techniques, a sinusoidal component in otherwise random data will appear as a sharp peak which might be confused with narrowband random contributions. A periodic or almost periodic signal when it is not a pure sinusoid, will cause harmonic peaks. But harmonic peaks of the narrowband random type can also appear in nonlinear systems. Hence, there are certain cases, in particular when sharp peaks appear at unexpected frequencies, where it is desirable to identify the presence of periodic components so they will not be misinterpreted as narrowband random components, and vice versa. There are several known methods for the peak type identification. They are concerned rather with the detection of sinusoids and suffer from severe drawbacks. (1) When a periodic signal wave is hidden in random noise, the signal recovery method can be applied if a synchronous trigger signal is available. Ordinarily, in reactor noise analysis applications such a trigger signal will not be available. (2) For the PSD estimation by the finite Fourier transform an appropriate window function or analyzer filter must be used to reduce leakage and the production of side lobes. If the measured spectral peak represents a sine wave, the indicated bandwidth will always be equal to the bandwidth of the analyzer filter, no matter how narrow the filter. The method of varying the bandwidth of the analyzer filter will clearly not work unless the resolution bandwidth of the analyzer filter is smaller than the bandwidth of possible narrowband random data (Bendat and Piersol, 1971). (3) The presence of sinusoidal components may also be revealed by an amplitude probability density function (PDF) estimation on single peaks isolated by filtering the data. The PDF plots for sinusoidal and random data are markedly different. However, the presence of the sine wave is difficult to identify if the variance of the sine wave is less than the variance of the random portion of the data (Bendat and Piersol, 1971). (4) ARMA modelling is useful for the exact determination of the frequency of sinusoids disturbed by rather small random noise provided one can assume that the peaks are due to periodic components (Kay and Marple, 1981; Kay, 1984; Vellante and Villante, 1984; Sakai, 1986).

A novel method called displaced spectra techniques is presented which allows under certain limitations to distinguish between the two peak types. It is based on Fourier transform techniques and requires the usual assumptions that the noise record to be analysed is stationary and of sufficient length, and that the peaks of interest are well separate and sufficiently intense in the PSD. Under these conditions the method should be suitable to be utilized as a simple routine procedure.

2. PRINCIPLES OF THE DISPLACED SPECTRA TECHNIQUES

For simplicity and brevity we limit our mathematical presentation to continuous functions in time and frequency. The usual digital approach would rather confuse the main issues which we are trying to present here. We introduce a new spectral function which is called here as second order displaced power spectral density (SDPSD) function. The SDPSD function is defined as the expected value to be taken on the segmented stationary noise signal y(t) (with the DC-component removed) by

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where $f$ is the frequency, $T$ is the time span of the signal segment, and $Y_k(f), k = 1, \ldots, 4,$ are instantaneous spectra following from
\[ Y_k(f) = \int_0^T w(t_k) y[n(k-1)T + t_k] e^{-2i\pi ft_k} dt_k \]

$w(t)$ is a window function which in particular we call here a signal window function in order to distinguish it from the later introduced correlation function window function. $w(t)$ is a positive function defined in the interval $(0,T)$ and is symmetric around $T/2$, i.e. $w(T-t) = w(t)$. The brackets in equation (1) denote ensemble averaging. For $\nu = 1$, the 4 signal segments are adjacent. If $\nu = 2$, there is a time gap of length $T$ between the segments. (It is convenient to consider the length of the time gap in units of $T$). For $\nu = 0$, the SDPSD function reduces to the second order power spectral density (SPSD) function which is equivalent to the 4th-order spectral moment. This case is of no interest here.

Overlap techniques can be applied to the estimation of the SDPSD function. We used the simple procedure to obtain a SDPSD estimate as the arithmetic average from a given number of subsequent instantaneous SDPSDs shifted by one $T$.

The principle of the method is a purification procedure. It makes use of the facts that the autocorrelation (AC) function of random noise always decays whereas the AC function of a sine wave oscillates infinitely in time. The SDPSD function has the following features: it shows a real-valued sharp peak like in the PSD from a sinusoidal component in $y(t)$, being independent of the initial phase (asynchronous run) and the delay times between the segments. It removes broadband random noise components in $y(t)$. It reduces significantly independent narrowband random noise components in $y(t)$. The degree of reduction depends on $T$, $\nu$ in the estimation procedure and the AC function of the narrowband random noise. Practical investigations showed that values $\nu > 2$ do not give appreciable improvements in the purification procedure.

3. THEORETICAL BACKGROUND

In order to get insight into the mechanism of the method we will theoretically treat the case that the record $y(t)$ consists of a sine wave $a(t)$ with the amplitude $A$ and the frequency $f_0$ and an additive Gaussian random noise $\xi(t)$.

\[ y(t) = a(t) + \xi(t) \]

Furthermore, we will show that under these assumptions the SDPSD function is real-valued. It is sufficient to perform the calculation for $\nu = 1$ (adjacent segments). Equation (1) can be derived from a 4th-order AC function by

\[ SDPSD(f) = \frac{1}{T^4} \times \int_0^T \int_0^T \int_0^T \int_0^T w(t_1)w(t_2)w(t_3)w(t_4) (y(t_1)y(T+t_2)y(T+t_3)y(T+t_4)) e^{2i\pi f(t_1-t_2+t_3-t_4)} dt_1 dt_2 dt_3 dt_4 \]

The numerically important contributions are obtained from a sum of products of double integral terms. Such a double integral term has the form

\[ \hat{S}(f,nT) = \frac{1}{T} \times \int_0^T w(t)w(t')R(nT + t - t')e^{-2i\pi f(t-t')} dt dt' \]

where $n$ is an integer number and $R$ represents an AC function (or an additive term in the 4th-order AC function). Equation (5) defines an expected PSD via the Fourier transform on a displaced AC function. It can be reduced to the single integral representation:

\[ \hat{S}(f,nT) = \int_{-T}^T W(\tau)R(nT + \tau) e^{-2i\pi f\tau} d\tau \]

$W(\tau)$ is a symmetric correlation function window function which results from

\[ W(\tau) = \frac{1}{T} \times \int_0^{T-|\tau|} w(t)w(t+|\tau|) dt \]

and extends over the range $-T \leq \tau \leq T$. (E.g. a rectangular signal window function leads to the Bartlett correlation function window function).

3.1 Case $y(t) = a(t)$

The required 4th-order autocorrelation function of the sine wave is given by

\[ (a(t_1)a(T+t_2)a(2T+t_3)a(3T+t_4)) = \]

\[ \frac{A^4}{8} \times \{ \cos 2\pi f_0 [(t_2 - t_1) - (t_4 - t_3)] + \cos 2\pi f_0 [2T + (t_2 - t_1) + (t_4 - t_3)] + \cos 2\pi f_0 [4T + (t_3 - t_1) + (t_4 - t_2)] \} \]
The resultant $SDPSD$ function follows from

$$SDPSD_a(f) = \frac{A^4}{16} \times \left( W^2(f - f_0) + W^2(f + f_0) \right)$$

(9)

where $W(f)$ is the (real-valued) Fourier transform of $W(\tau)$ and is assumed to be sharply peaked at $f = 0$.

3.2 Case $y(t) = z(t)$

The 4th-order autocorrelation function of Gaussian random noise can be decomposed into a sum of products of ordinary AC functions.

$$\langle z(t_1)z(T + t_2)z(2T + t_3)z(3T + t_4) \rangle =$$

$$R_z(T + t_2 - t_1)R_z(T + t_4 - t_3) + R_z(3T + t_4 - t_1)R_z(T + t_3 - t_2)$$

(10)

According to this decomposition the $SDPSD$ function consists of a sum of three terms, the first two terms resulting from the use of equation (6):

$$SDPSD_a(f) = \left| \hat{S}_z(f, T) \right|^2 + \left| \hat{S}_z(f, 2T) \right|^2 + \varepsilon_a(f)$$

(11)

The last term, $\varepsilon_a(f)$, has been found to be almost negligibly small. Its explicit form and the proof that it is a real function of frequency is given in appendix A. The most important term is the first term in equation (11). It determines the degree of purification, and in the case of narrowband random noise, the reduction in the peak power. It is to note that for $\nu = 0$, equation (11) reduces to the well known relationship that the $SPSD$ is twice the square of the $PSD$. This can easily be derived from the Rice representation of Gaussian random noise. In this case, $\varepsilon_a(f)$ presents a bias term.

3.3 General case $y(t) = a(t) + z(t)$

In this general case a lot of cross-terms appears. According to the theorem of Wang-Uhlenbeck (1945) all odd order correlations of the Gaussian random noise average to zero. The resulting $SDPSD$ function consists of a sum of 8 terms where the first two terms follow from the equations (9) and (11).

$$SDPSD_{a+z}(f) = SDPSD_a(f) + SDPSD_z(f) + \hat{S}_a(f, T) \hat{S}_z(f, T) + \hat{S}_z(f, 2T) \hat{S}_a(f, 2T) + \varepsilon_{aa}(f) + \varepsilon_{aa}(f)$$

(12)

$\hat{S}_a(f, T)$ and $\hat{S}_a(f, 2T)$ respectively are obtained from equation (6) with the AC function of the sine wave. The functions $\varepsilon_{aa}(f)$ and $\varepsilon_{aa}(f)$ are similar to $\varepsilon_a(f)$ as given in the appendix A. The difference is only that in the product of the entering AC functions the one belongs to the sine wave and the other one to the random noise component. Since no use of special forms of these AC functions is made, $\varepsilon_{aa}(f)$ and $\varepsilon_{aa}(f)$ are also real functions. Hence, from the form of equation (12) we can infer that under the made assumptions the $SDPSD$ function is a real-valued function. Practical test examples showed that with increasing number of averages the imaginary part of the estimated $SDPSD$ function tends to converge to zero. They suggested to use only the real part in estimated $SDPSD$ functions and to disregard any residuals of the imaginary part.

If $z(t)$ represents broadband background noise, all terms in equation (12) with the exception of the first one will practically vanish, thus that

$$SDPSD_{a+z}(f) \approx SDPSD_a(f)$$

(13)

A similar calculation can be made assuming that $a(t)$ represents independent Gaussian narrowband random noise. Because the AC function of narrowband random noise decays markedly slower than that of the broadband background noise, equation (13) will exhibit certain rest part contributions from such a peak. In order to distinguish between the two peak types, a validation criterion is required. The $SPSD$ function (or the $SDPSD$ function for $\nu = 0$ respectively) is not a suitable reference function, when one modifies equation (12) correspondingly, because then the cross-terms come up. Independent processes are not separable in this function. But they are separable in the $PSD$.

4. VALIDATION CRITERION

Several attempts have been made to establish a criterion for the peak type validation. It should be unaffected by bias errors arising in the digital estimation approach. As presently one of the best, we found the peak power ratio (hereafter called as the $\xi - ratio$), defined by

\footnote{We do not consider the independence assumption as a constraint since the presence of peaks in otherwise flat background noise indicates the presence of another process.}
\[ \xi = \frac{\int_{-\infty}^{\infty} \text{sign} \left[ \text{Real} (SDPSD(f)) \right] \sqrt{\text{Real} (SDPSD(f))} \, df}{\int_{-\infty}^{\infty} SDPSD(f) \, df} \]  

The integrals extend over the peak areas with subtracted background in the PSD function and SDPSD function. For a pure sinusoid without the presence of background noise this ratio is always exactly equal to 1. In the digital estimation procedure of the PSD function and the SDPSD function by Fast Fourier Transform (FFT) techniques we always used a segment size of 1024 data points to get good peak resolution. The influence of several signal window functions, like the Hanning window, the Blackman-Harris 4-term (-74db) window and the Kaiser-Bessel window with the parameter \( \alpha = 4 \) (Harris, 1978) was investigated. The latter two windows can be regarded as being equivalent, but broaden appreciably a sinusoidal peak against its estimation through the Hanning window. We finally adopted the Hanning signal window because it showed greater advantage with regard to the background subtraction procedure. It should be noted that the resultant spectral window function \( W(f) \) has positive side lobes only with the highest side lobe level at -31.5 db. A plot of this function is given in the paper of Nuttall (1981). The background below a peak was estimated in a semiautomatic way by a linear least squares fit using 6 points on the left hand side and on the right hand side respectively of the detected peak.

![Graph showing standard deviation as a function of SNR.](image)

**Fig. 1.** Estimated standard deviation \( \sigma_\xi \) for a sinusoid as function of the signal to noise ratio SNR.

The identification method is based on the requirement that broadband random noise is removed by the SDPSD function. In practice, the purification from background noise will not be ideal. We found that more consistent \( \xi - \text{ratio} \) data are obtained if background residuals are subtracted in the SDPSD in the same way as in the PSD. However, the most important influence on the scattering of \( \xi \) comes from the background amplitude in the PSD and its estimation accuracy (number of averages). This scattering error involves the most important limitation in the application of the method. Fig. 1 shows plots of the standard deviation \( \sigma_\xi \) of \( \xi \), with \( \nu = 1 \) for the SDPSD, as function of the signal to noise ratio (SNR, expressed in db-units) for different number of averages in the PSD estimation (3 averages less in the corresponding SDPSD estimation). The SNR is defined here as the ratio of the peak power in the PSD to the local background noise power below the peak, its shape being approximated by a triangle. The data were obtained from computer simulated Gaussian broadband noise records and a sine wave additively mixed in. Each data point shown in Fig. 1 is based on a \( \sigma_\xi \) estimation on about 50 \( \xi \)-determinations repeated under independent statistical conditions (different record sections). In each case the average value of \( \xi \) was closely to 1. The \( \xi - \text{ratio} \) distribution function was checked at \( SNR = \pm 1 \text{ db} \) for average numbers \( \geq 50 \) and was found to be normal with \( \left( 1, \sigma_\xi \right) \left( \chi^2\text{-goodness} - \text{of-fit test on 5 \% level of significance} \right) \). The figure exhibits the well expected trend that a reliable \( \xi \)-determination requires an increasing accuracy in the PSD estimation for decreasing SNRs.

In the analysis of rather long noise records, a present deterministic periodic component can often not be presumed as being ideally stationary. We investigated the influence of two cases of nonstationary sinusoids on \( \xi \). In the first case we gave a trend of a continuous linear amplitude change over the time of the total record length. The frequency \( f_0 \) was kept constant. In the second case, the amplitude was kept constant, but the frequency was continuously linearly changed. This leads to a peak broadening which gives the appearance of the picture of a narrowband random noise process, in particular if such a change remains within a small percentage and background noise is present. In both cases, with changes up to 10\%, the \( \xi - \text{ratio} \) was practically not affected by these trends and equals very closely to 1.

For the study of the \( \xi - \text{ratio} \) behaviour on narrowband random data we selected a second order resonance filter given in the textbook of Moschytz and Horn (1981). This filter is shown in Fig. 2. For equal resistances \( R_1 = R_2 = R_3 = R \)
and equal capacitor values \( C_1 = C_2 = C \), the analysis of the circuit leads to the Laplace transformed transfer function
\[
H(s) = \frac{2}{Q\left(\frac{1}{Q} + \frac{s}{\omega_0} + \frac{\omega_0}{s}\right)}
\]
where the resonance frequency \( \omega_0 \) (in radians per sec) is given by \( \omega_0 = 1/\sqrt{R_4 R C} \) and the quality factor \( Q \) follows from \( Q \equiv \omega_0/\Delta \omega = R_3/\sqrt{R_4 R} \). \( \Delta \omega \) is the half-power bandwidth. \( Q \) must be larger than 0.5. For white noise input with the input \( PSD = q^2 \), the output signal has the AC function
\[
R_e(\tau) = \frac{2\omega_0 q^2}{Q \sin \eta} e^{-\omega_0 |\tau|} \cos \eta \sin(\eta - \omega_0 |\tau| \sin \eta)
\]
where \( \cos \eta = 1/2Q \). Equation (16) has been used to calculate \( SDPSD \) functions and to derive theoretical \( \xi - ratios \)

![Diagram of the second order resonance filter.](image)

**Fig. 2.** Diagram of the second order resonance filter.

as a function of \( Q \). In the example given in Fig.3, a resonance frequency at 4Hz and a Nyquist cutoff frequency of 12.8 Hz are assumed. In the calculations the Hanning signal window function has again been used. The points given in this figure are estimates obtained from computer simulated noise records (without addition of background noise). The noise generation procedure has been based on a (2,2) ARMA model with independent Gaussian random data as input and by applying the bilinear z-transform (e.g. Lacroix, 1980) to equation (15) with correspondingly transformed data of \( \omega_0 \) and \( Q \) into the pseudo-frequency region.

![Graphs showing \( \xi - ratios \) as a function of \( Q \)-value.](image)

**Fig. 3.** \( \xi - ratio \) of narrowband random noise as function of the \( Q \)-value. \( (f_0 = 4 \text{ Hz, } T = 40 \text{ sec.}) \)

The \( \xi - ratio \) has an upper theoretical limit of \( \approx \sqrt{2} \) for random noise. On the other hand, if \( Q \to \infty \), the narrowband random noise components die in the case of our special resonance circuit. For very high \( Q \)-values, there is no possibility
of distinguishing the peak types by the $\xi - ratio$. We consider Q-values of up to 500 which may occur in practical damped systems, as a realistic range. Since in the estimation procedure used by us, the shape of a narrowband random noise peak in the PSD becomes indistinguishable from that of a sinusoid for $Q > 50$, the interesting application range of the method may be for Q-values from 50 to 500. In this range the $\xi - ratio$ is well below 1 for narrowband random noise. We have not made similar extensive investigations on the scattering of $\xi$ and the influence of background noise to establish the criterion on a precise confidence statement. Our computer simulation experiments showed that in the most interesting case when appreciable background noise is present, a peak can well be identified as due to a narrowband random process if the $\xi - ratio$ is significantly below the scattering range $\sigma_\xi$ for a sinusoidal peak (Fig.1). As a measure we presently accepted the simple decision criterion $\xi < 1 - 2 \sigma_\xi$ with $\sigma_\xi < 0.2$. The latter condition might be considered to define the working range of the method. On the other hand, a peak is accepted as sinusoidal if $\xi > 1 - \sigma_\xi$. For values $1 - 2 \sigma_\xi \leq \xi \leq 1 - \sigma_\xi$, no interpretation can be given.

5. EXAMPLES

5.1 Analogue simulation experiment

Meier (1987) has investigated the practical layout of the resonance filter given in Fig.2 for obtaining high Q-values. If one uses frequency-compensated operational amplifiers, the critical components for resonance frequencies below 1 KHz are the capacitors which must have very small loss factors. He realized a filter for $f_0 = 40$ Hz and Q-values of up to 500. For testing the filter, the step pulse response function was used. The measured Q-values deviated from the designed ones by less than $\pm 5\%$.

An experimental circuit was made up in the 100 Hz frequency region for the generation of stationary analogue noise records which contain Gaussian broadband random noise components, sinusoidal components and Gaussian narrowband random noise contributions including a squared portion. The circuit used is similar to that one given in a recent paper by Behringer and Pifeyro (1986). Four signals were generated and mixed together by a summer. The broadband random noise signal was obtained from a Gaussian white noise generator and a low-pass filter. Sinusoids were generated by periodic rectangular pulses (with < 20% duty cycle) from a pulse generator. The input signal to the resonance filter was taken from a second independent Gaussian white noise generator. The filter output signal was branched to a squarer which gives non-Gaussian narrowband random noise contributions. The mixed signal was high-pass filtered to remove the DC-component and a sharp antialiasing filter was inserted before the AD-converter.

<table>
<thead>
<tr>
<th>Peak No</th>
<th>Peak type</th>
<th>$f_0$ (Hz)</th>
<th>Case 1, Q = 103</th>
<th>Case 2, Q = 295</th>
<th>Case 3, Q = 505</th>
</tr>
</thead>
<tbody>
<tr>
<td>1</td>
<td>sinusoidal, fundamental</td>
<td>16.4</td>
<td>1.02</td>
<td>-1.1</td>
<td>1.01</td>
</tr>
<tr>
<td>2</td>
<td>sinusoidal, 1st harmonic</td>
<td>32.9</td>
<td>1.01</td>
<td>-2.3</td>
<td>0.99</td>
</tr>
<tr>
<td>3</td>
<td>narrowband</td>
<td>40.0</td>
<td>0.06</td>
<td>0.05</td>
<td>0.13</td>
</tr>
<tr>
<td>4</td>
<td>sinusoidal, 2nd harmonic</td>
<td>49.3</td>
<td>0.77</td>
<td>-2.5</td>
<td>0.43</td>
</tr>
<tr>
<td>5</td>
<td>narrowband, squared signal</td>
<td>80.0</td>
<td>0.006</td>
<td>0.00</td>
<td>0.03</td>
</tr>
</tbody>
</table>

Table 1 represents analysis results of 3 cases with different Q-values of the resonance filter. Case 1 differs from the other cases by slightly different signal mixing ratios, and also by a somewhat smaller duty cycle in the rectangular pulse signal. The sampling frequency was 208 Hz. Each PSD function was estimated by 100 averages. The PSD function and the $SDPSD$ function of case 2 are shown in Fig.4. On 5 peaks, $\xi - ratios$, denoted by $\xi_n$, could be determined. The 3rd harmonic peak of the periodic signal contribution has a too small SNR. The 4th harmonic peak which is expected to appear very near above the peak of the squared narrowband random signal contributions disappears completely in the background noise. For comparison, theoretical $\xi - ratios$, denoted by $\xi_n$, have been calculated for the narrowband random noise contributions. The AC function required in the calculation for the squared contributions follows from $R_{\xi n}(\tau) = 2R_{\xi n}(\tau)$. There is in general a good agreement between $\xi_n$ and $\xi_n$, except the data obtained on peak 4. If this peak would be the only in the spectrum and nothing would be known about is origin, the identification possibility becomes doubtful.
Our method is in general not suitable for the identification of higher harmonics. If the type of a peak at the fundamental frequency can be well identified by this method, the complementary method for harmonic analysis is the bispectrum (Hasselmann et al., 1983) or the 'criss'-spectrum techniques (Väth, 1979).

5.2 Neutron noise

A relatively short noise record was available from one of the neutron chambers of the control instrumentation at our swimming pool reactor SAPHIR where the reactor was operated with maximum coolant flow rate. This flow rate is about twice than that one under normal operation conditions. In the PSD function two sharp peaks at \( f_0 = 0.82 \) Hz and 2.92 Hz respectively appeared. It is obvious that mechanical core vibrations induced by the coolant flow are expected to be of random nature. We reanalysed this record for testing our method on actual noise. In Fig.5 the PSD function and the SDPSD function (for \( \nu = 1 \)) are shown. The PSD estimation is based on 25 averages. This number is sufficient for the peak type identification since the SNR is in the order of 10 db for both peaks. The obtained \( \xi - \) ratios are \( \xi_{peak1} \approx 0.02 \) and \( \xi_{peak2} \approx 0.1 \). These values clearly characterize the peaks to be of the narrowband random type. The example shows a further feature of the SDPSD function. Peak 1 seems to contain a small unresolved contribution from a nearby other peak in the PSD function. This contribution appears more clearly in the SDPSD function (but with negative sign). Computer simulation experiments showed that the SDPSD function can be helpful for visualising fine structures in a peak which are unresolved in the PSD function. This feature, however, requires further investigation.

Fig. 4. PSD function and SDPSD function (\( \nu = 1 \)), case 2 of the simulation experiment.

Fig. 5. PSD function and SDPSD function (\( \nu = 1 \)), neutron noise from reactor SAPHIR.

6. CONCLUDING REMARKS

For noise data qualification the displaced spectra techniques are believed to be a useful tool for distinguishing between sinusoidal components and narrowband random noise contributions in otherwise random noise data. The computation procedure of the SDPSD function can easily be added to any code for PSD estimation based on FFT-techniques. The proposed validation criterion requires the subtraction of the background below the interesting peaks. In order to make
the method more suitable for routine applications we started to develop a method which undertakes this background subtraction in a semiautomated way with the support of a graphical terminal (and available graphical software). There are limitations in the peak identification. (1) Peaks of interest must be well separate from neighbouring peaks. (2) Very narrowband random noise contributions (as far as they should occur with sufficient intensity) cannot be distinguished from a sinusoid. (3) A high background portion below the peak of interest requires a large number of averages in the estimation of the spectral functions. From the practical point of view, the working range of the method may not extend to SNRs below -3 dB. A reduction of the statistical scattering of the $\xi$–ratio can be attained if one would apply the overlap method of Welch (1967) or the more recently lag reshaping method of Nuttall and Carter (1982) to the estimation of the spectral functions in order to retrieve the loss in degrees of freedom due to windowing. (4) A further limitation is introduced by the decision criterion for the $\xi$–ratio which is presently based on simple engineering trials.

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REFERENCES


APPENDIX A: Explicit Form of the Term $\varepsilon_a(f)$

The term $\varepsilon_a(f)$ appearing in equation (11) has the explicit form:

$$\varepsilon_a(f) = T_1(f) \times T_2(f)$$  \hspace{1cm} (A-1)

where $T_1(f)$ and $T_2(f)$ are given by

$$T_1(f) = \frac{1}{T} \times \int_0^T dt_1 dt_2 w(t_1) w(t_2) R_a(3T + t_2 - t_1)e^{2\pi if(t_1 + t_2)}$$  \hspace{1cm} (A-2)

$$T_2(f) = \frac{1}{T} \times \int_0^T dt_2 dt_3 w(t_2) w(t_3) R_a(T + t_3 - t_2)e^{2\pi if(t_2 + t_3)}$$  \hspace{1cm} (A-3)

When one introduces the notations

$$F_1(\tau) = R_a(3T - \tau) + R_a(3T + \tau)$$  \hspace{1cm} (A-4)

$$F_2(\tau) = R_a(T - \tau) + R_a(T + \tau)$$  \hspace{1cm} (A-5)

equations (A-2) and (A-3) can be written as

$$T_k(f) = \int_0^T d\tau F_k(\tau)e^{2\pi if\tau} \left[ \frac{1}{T} \int_0^T dt w(t)w(t+\tau)e^{2\pi if\tau} \right], \quad k = 1, 2$$  \hspace{1cm} (A-6)

The signal window function $w(t)$, defined in the interval $0 \leq t \leq T$, can be expressed as a Fourier series (Nuttall, 1981):

$$w(t) = \sum_{n=-\infty}^{\infty} a_n \exp \left( \frac{2\pi iT}{T} \right)$$  \hspace{1cm} (A-7)

with real coefficients $a_{n}$. An important class of windows is represented by a sum of a limited (small) number of terms (Harris, 1978). The Hanning window, the Hamming window and the family of the Blackman windows belong to this class. The use of equation (A-7) allows the representation of the equation (A-6) by
Displaced spectra techniques

\[ I_k(f) = \sum_{\nu,\mu} \frac{a_{\nu}a_{\mu}}{2\pi i(2fT + \nu + \mu)} \left[ e^{i\pi fT} (C_{\nu\mu}(f) - iQ_{\nu\mu}(f)) - (C_{\nu\mu}(f) + iQ_{\nu\mu}(f)) \right] \quad (A - 8) \]

with

\[ C_{\nu\mu} = \int_{0}^{T} d\tau F_k(\tau) \cos \left[ 2\pi (f + \frac{\nu}{T}) \tau \right] \quad (A - 9) \]
\[ Q_{\nu\mu} = \int_{0}^{T} d\tau F_k(\tau) \sin \left[ 2\pi (f + \frac{\nu}{T}) \tau \right] \quad (A - 10) \]

If one finally inserts equation (A-8) into the equation (A-1), a real-valued function \( \varepsilon_\omega(f) \) results.

\[ \varepsilon_\omega(f) = \sum_{\nu,\mu} \frac{a_{\nu}a_{\mu}a_{\nu'}a_{\mu'}}{\pi^2(2fT + \nu + \mu)(2fT + \nu' + \mu')} \times \]
\[ (C_{1\nu}(f) \sin 2\pi fT - Q_{1\nu}(f) \cos 2\pi fT) \times (C_{2\nu'}(f) \sin 2\pi fT - Q_{2\nu'}(f) \cos 2\pi fT) \quad (A - 11) \]

One can show that for frequency values which lead to a zero-denominator, the corresponding term in the sum of equation (A-11) converges to a well defined limit.
A RANDOM WALK TECHNIQUE TO ANALYSE NOISE AND DETECT MALFUNCTION IN A REACTOR

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Abstract - A new statistical decision algorithm to detect malfunction in an operating reactor by noise analysis is presented in this paper. The algorithm is based on Random Walk and First Passage Time formulations. We apply this method to detect onset of boiling in a benchmark experiment carried out at KNS-I loop. The paper presents the details of the method and discusses its performance on the benchmark experiment.

1. INTRODUCTION

Noise analysis has become a viable and attractive technique for detection and diagnosis of malfunction in an operating reactor system. The chief merit of these techniques is that they are non-intrusive and do not impose any constraints on the reactor operation. For a good account of the various noise analysis techniques see Williams (1976) and for some recent investigations in this field see Thie (1981). The identification of a suitable pattern and employing it a statistical decision algorithm is an important area in noise analysis and there have been numerous studies reported in the recent times see Williams and Cormick (1987).

In this paper, we propose a statistical decision algorithm based on Random Walk and First Passage Time formulations. We describe in Section 2 the theoretical basis of this formulation. In Section 3, we illustrate the use of the technique to detect boiling in a Liquid Metal cooled Fast Breeder Reactor (LMFBR). Specifically, we apply the technique to analyse KNS-I experiments and study its performance with the measurements. In Section 4, we discuss the results and in Section 5, we summarise the principal conclusions of the study.

2. FORMULATION OF THE STATISTICAL DECISION ALGORITHM

When the reactor is operating in its normal state, there is a steady state fluctuations in a macroscopic parameter which we term as noise. It could be neutron noise, acoustic noise, temperature fluctuations, etc. Let \( x(t) \) formally denote the stochastic process that represents, say, acoustic noise, when the reactor is under normal operation. When there is a malfunction (for example, coolant boiling due to blockage in a channel) the statistics of the noise changes; let \( y(t) \) represent the noise under abnormal operation (or operation with malfunction). One of the aims of noise analysis technique is to find the time of onset of the malfunction by detecting the change in the noise from \( x(t) \) to \( y(t) \). Usually, one obtains statistical descriptors of the noise like mean, root mean square deviation, skewness, kurtosis, correlation, power spectra, etc., and detect
malfunction by looking for any statistically significant changes in one or many of these descriptors employing suitable statistical decision algorithms. Accordingly, the first step in our formulation consists of identifying a suitable statistical descriptor and we demand of it the following characteristics.

We discretise the time axis as \( t_n = t_0 + n \Delta t \), \( n = 0, 1, \ldots \) where \( t_0 \) is an arbitrary initial time. We define a random variable \( s'_i \) such that it takes random and independent values in the time segments defined above. Let \( s'_i \) formally denote the value of \( s'_i \) in the \( i \)-th time segment. The value of \( \Delta t \) is so chosen that \( s'_i \) is independent of \( s'_j \) for all \( j \neq i \). Let \( f(s'_i) \) denote the probability density of \( s'_i \) when the reactor is under normal operation, \( x(t) \) being the corresponding noise and \( g(s'_i) \) be the probability density of \( s'_i \) under abnormal operation, \( y(t) \) being the corresponding noise signal. We assume \( f(s'_i) \) and \( g(s'_i) \) to be known. In practice, however, it would be difficult to know or learn the density functions \( f \) and \( g \). Hence we model \( f \) and \( g \) to be Gaussian with their means and standard deviations obtained apriori or learnt. Let \( m_1 \) and \( \sigma_1 \), \( m_2 \), \( \sigma_2 \) denote the means and standard deviation of \( f \) and \( m_2 \), \( \sigma_2 \) the corresponding quantities for \( g \). Also, we select the feature random variable \( z'_i \) such that the two densities are reasonably well separated with, say \( m_1 < m_2 \). Let us identify an optimal point \( z'_i \), called the threshold such that \( m_1 < z'_i < m_2 \). Indeed we shift the origin to \( z'_i \) and in the new coordinate system \( f(z'_i) \) and \( g(z'_i) \) have their means at \( \mu_1 = m_1 - z'_i \) and \( \mu_2 = m_2 - z'_i \) respectively.

The statistical decision algorithm based on the feature \( z'_i \) is as follows. We analyse the noise from the reactor and calculate the value of the feature \( z'_i \) in the successive time segments. We obtain, the feature \( z'_i \) in the successive time segments. We define a continuous random walk in discrete time as

\[
S_i = S_{i-1} + z'_i, \quad i = 1, 2, \ldots \quad j \quad S_0 = 0
\]

We set two absorbing boundaries on the real line, one at \( - \alpha \) (called left boundary) and the other at \( \beta \) (called right boundary), with \( \alpha, \beta > 0 \). The course of the random walk is thus determined by the successive values of \( S_i \) obtained from the analysis of the noise. When the walk crosses the left barrier, we take a decision that the reactor is under normal operation, reset the walk to origin and continue. When the walk crosses the right barrier, we take a decision that there is malfunction in the reactor. Again we reset the walk to origin and continue.

Since the decision algorithm is statistical there is a non-zero probability for the decision to be wrong. This, in other words mean the following. When the reactor is under normal operation there is a non-zero probability for the walk to cross the right barrier and lead to a spurious decision. Let \( P_s \) denote the probability for a spurious decision. In the same way, when the reactor is operating with a malfunction, there is a non-zero probability for the walk to cross the left barrier leading to a wrong decision. Let \( P_m \) denote the probability of missing a malfunction signal. Evidently, \( P_s \) and \( P_m \) are related to the densities \( f \) and \( g \) and to the barrier parameters \( \alpha \) and \( \beta \). These relations are derived in the following.

Consider the random walk defined by Eq.(1), generated by a gaussian,

\[
G(z) = (2\pi \sigma^2)^{-1/2} \exp \left[ -(z - \mu)^2 / 2\sigma^2 \right]
\]

of mean \( \mu \) and variance \( \sigma^2 \). The moment generating function of the gaussian random variable is given by
$$m(\theta) = \int_{-\infty}^{\infty} \exp(\theta \xi) G(\xi) d\xi = \exp(\mu \theta + \sigma^2 \theta^2/2)$$  \hspace{1cm} (3)

Let \( A \) denote the ensemble of all random walks defined by Eq.(1) on the real line. Let us denote by \( N \), a random variable called First Passage Time, defined on \( A \) and \( N \) is the number of steps a random walk takes to cross first the barrier (left or right). Let \( SN \) be the random variable denoting the position of the random walk after \( N \) steps. We introduce an identity due to Wald (See Bartlett, 1966) that relates \( N \), \( SN \) and \( m(\theta) \). It is given by

$$\langle \exp(\theta SN) \rangle \mathbb{E}[-m(\theta)]^{-N} = 1$$  \hspace{1cm} (4)

where the angular bracket denotes an average over \( A \). The above identity is valid for all values of \( \theta \) such that \( |m(\theta)| > 1 \). From Eq.(3) it is clear that \( m(\theta) \to \infty \) as \( \theta \to \pm \infty \) and, the equation \( m(\theta) = 1 \) has two real roots (when \( \mu \neq 0 \)) one at \( \theta_1 = 0 \) and the other at \( \theta_2 = -2\mu/\sigma^2 \). Let us consider a situation when \( |\mu| \) and \( \sigma \) are small compared to \( a \) and \( b \). Then we can write the Wald identity for \( \theta_1 \) and \( \theta_2 \) and get

$$P_a + P_b = 1$$  \hspace{1cm} (5a)

$$e^{-\theta_1 a} P_a + e^{-\theta_2 b} P_b = 1$$  \hspace{1cm} (5b)

where \( P_a \) and \( P_b \) are the probability for the walk to cross the left and right barrier respectively. Eq.(5) above can be solved to yield \( P_a \) and \( P_b \).

The probability for a spurious decision, \( P_S \) is given by \( P_b \) when \( \mu = \mu_1 \) and \( \sigma = \sigma_1 \). Explicitly, it is given by

$$P_S = \frac{e^{2\mu_1 a/\sigma_1^2} - 1}{e^{2\mu_1 a/\sigma_1^2} - e^{-2\mu_1 b/\sigma_1^2}}$$  \hspace{1cm} (6)

Similarly, \( P_m \) - the probability of missing a malfunction signal is given by \( P_a \) when \( \mu = \mu_2 \) and \( \sigma = \sigma_2 \). Explicitly, \( P_m \) is given by

$$P_m = \frac{e^{-2\mu_2 b/\sigma_2^2} - 1}{e^{-2\mu_2 b/\sigma_2^2} - e^{2\mu_2 a/\sigma_2^2}}$$  \hspace{1cm} (7)

Any statistical decision algorithm should meet specified criterion on \( P_S \) and \( P_m \). For example, for coolant boiling detection, the criterion is that the probability of missing a boiling signal should not exceed \( 10^{-4} \) and there should not be more than one spurious signal in 10 years of continuous operation. In the algorithm presented here, the time taken per decision is a random variable, \( T \) given by \( T = N \Delta t \). Under normal operation the mean time per decision is

$$\langle T \rangle = \frac{a(1-P_S) + b P_S}{P_S}$$

\hspace{1cm} (8)

Thus, the average number of decisions that one would take in 10 years under normal reactor operation is \( 3.1536 \times 10^8 / \langle T \rangle \). Thus the criterion for spurious decision can be stated as that \( P_S \) should not exceed \( 3.17 \times 10^{-7} / \langle T \rangle \). Let us denote \( P_S^C \) and \( P_m^C \) the specifications on \( P_S \) and \( P_m \). For the example considered above \( P_S^C = 3.17 \times 10^{-7}/ \langle T \rangle \) and \( P_m^C = 10^{-3} \). We systematically search for the values of the barriers \( a \) and \( b \) such that \( P_S < P_S^C \) and \( P_m < P_m^C \).

Thus with the selection of \( a \) and \( b \) on the basis of \( P_S^C \) and \( P_m^C \) the statistical decision algorithm is complete and in the next section we shall
illustrate the performance of the algorithm by analysing KNS-1 signal.

3. DETECTION OF SODIUM BOILING IN LMPBRs

The above formalism has been applied to predict onset of boiling and the presence of boiling on a test data made available by IAEA for the coordinated research programme (CRP). The test data are from experiments on KNS-I loop (Hubber et al., 1976) of Kernforschungszenrum on KNS-I loop. These data were prepared at KFK (Rohrbacher and Aberli 1984) by mixing background flow noise and boiling noise. The supplied data are in the form of 12 analogue files. Each file consists of three regions, namely, flow noise, initial boiling noise and intense boiling noise. In successive files, the signal to noise ratio (S/N) is reduced by 2 dB so that the first file data with S/N = 0 dB and 12 th file data with S/N =-17.5 dB. The duration of the file is 105 sec and data are recorded with a tape speed of 30 inch per second.

The analogue data has been digitised using 12 bit analogue to digital converter (ADC). The data have been digitised at the intervals of 52 micro seconds and the tape was run at 1/16 of the original recording speed to preserve high frequency data. The data has been stored on digital tape. 2K data points are put as one record and forty such records are put as one digital file.

3.1. Feature Selection

Probability Density Functions (PDF) have been calculated by sorting out digital data into 64 equally spaced slots of pulse heights. Eighty digital files have been taken in the flow noise region and forty digital files have been taken in the intense boiling region to calculate PDF in flow noise and boiling noise regions respectively. Fig.1 gives the PDF distribution both in background and boiling regions.

![Fig.1. PDF of the Signals (File-1)](image)

From this figure we notice that PDF value is sharper in the background region compared to the boiling region and these two curves intersect at two points. Between these two points the area under the PDF curve is less in the boiling region compared to background region. Outside these two points the PDF values are more for boiling region compared to the background region. These three regions are selected as feature to discriminate boiling and background. Table 1 gives the mean values of these areas and their standard deviations, both in background and boiling regions for file 1 and 12.
Table 1
Left and Right Boundary of Various Features

<table>
<thead>
<tr>
<th>Feature</th>
<th>Background Region</th>
<th>Boiling Region</th>
<th>Threshold</th>
<th>Left Boundary</th>
<th>Right Boundary</th>
</tr>
</thead>
<tbody>
<tr>
<td></td>
<td>Mean Value ( m_1 )</td>
<td>Standard Deviation ( \sigma_1 )</td>
<td>Mean Value ( m_2 )</td>
<td>Standard Deviation ( \sigma_2 )</td>
<td></td>
</tr>
<tr>
<td>File 1</td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>A_1</td>
<td>0.742</td>
<td>1.241</td>
<td>44.2</td>
<td>34.5</td>
<td>4.008</td>
</tr>
<tr>
<td>A_2</td>
<td>1405.6</td>
<td>37.9</td>
<td>1025.2</td>
<td>89.2</td>
<td>1281.9</td>
</tr>
<tr>
<td>A_3</td>
<td>188.8</td>
<td>40.8</td>
<td>498.6</td>
<td>130.6</td>
<td>281.7</td>
</tr>
<tr>
<td>File 12</td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>A_1</td>
<td>0.110</td>
<td>0.538</td>
<td>1.221</td>
<td>1.631</td>
<td>0.777</td>
</tr>
<tr>
<td>A_2</td>
<td>1027.5</td>
<td>27.4</td>
<td>1009.4</td>
<td>27.2</td>
<td>1018.0</td>
</tr>
<tr>
<td>A_3</td>
<td>433.1</td>
<td>37.3</td>
<td>495.3</td>
<td>39.7</td>
<td>464.2</td>
</tr>
</tbody>
</table>

The threshold between boiling and background has been fixed as described by Reddy (1987) and Om Pal Singh et al. (1987). The thresholds of these feature are also given in the Table 1.

The target values recommended by International Atomic Energy Agency (IAEA) for Pm is \( 10^{-3} \) and for Ps it is such that spurious trips are less than one in ten years. For the random walk model described in the last section, left and right boundaries \( a \) and \( b \) are fixed, using Eq.(6) and (7), such that we meet target requirements for Ps and Pm. These boundaries are also given in Table 1 for file 1 and file 12.

4. RESULTS AND DISCUSSIONS

Using mean values, standard deviations, thresholds, left and right boundaries random walk has been performed using KNS-I experimental data, to predict time or the onset of boiling. Table 2 gives the times of onset of boiling predicted by different features and the reported true time of onset of boiling.

Table 2
Onset of Boiling Predicted by Different Methods

<table>
<thead>
<tr>
<th>Method</th>
<th>File 1</th>
<th>File 12</th>
</tr>
</thead>
<tbody>
<tr>
<td>Random Walk with</td>
<td></td>
<td></td>
</tr>
<tr>
<td>Feature ( A_1 )</td>
<td>26.8 s</td>
<td>23.8 s; 24.2 s; 28.5 s</td>
</tr>
<tr>
<td>( A_2 )</td>
<td>29.3 s</td>
<td>51.2 s</td>
</tr>
<tr>
<td>( A_3 )</td>
<td>34.0 s</td>
<td>34.1 s</td>
</tr>
<tr>
<td>True Time</td>
<td>27.0 s</td>
<td>27.8 s</td>
</tr>
</tbody>
</table>

For file 1, where signal to noise ratio is good, the onset of boiling has been predicted very well by feature 1 and feature 2, whereas for file 12 feature 1 gives boiling signals at 23.8; 24.2 and 28.5 secs. Feature 2 and 3 predict boiling quite later, on file 12.

In Table 3, we present mean theoretical time for decision making in background and boiling regions as calculated from Eq. (6).
Table 3
Theoretical and Experimental Average Times for Boiling and Non-boiling Decisions

<table>
<thead>
<tr>
<th>File No.</th>
<th>Feature</th>
<th>Background Region</th>
<th>Boiling Region</th>
</tr>
</thead>
<tbody>
<tr>
<td></td>
<td></td>
<td>Mean Theoretical Time</td>
<td>Mean Experimental Time</td>
</tr>
<tr>
<td>1</td>
<td>A₁</td>
<td>26.6</td>
<td>27.5</td>
</tr>
<tr>
<td></td>
<td>A₂</td>
<td>1.16</td>
<td>1.74</td>
</tr>
<tr>
<td></td>
<td>A₃</td>
<td>2.93</td>
<td>3.83</td>
</tr>
<tr>
<td>12</td>
<td>A₁</td>
<td>31.6</td>
<td>32.3</td>
</tr>
<tr>
<td></td>
<td>A₂</td>
<td>87.9</td>
<td>91.0</td>
</tr>
<tr>
<td></td>
<td>A₃</td>
<td>5.63</td>
<td>6.54</td>
</tr>
</tbody>
</table>

Time in terms of no. of records

Here, we also present the experimentally observed mean time for decision taking.

Table 4 and 5 give the number of boiling and non-boiling decisions taken by different features for file 1 and 12.

Table 4
Number of Boiling and Non-boiling Decisions for File 1

<table>
<thead>
<tr>
<th>Time in Sec.</th>
<th>Feature 1</th>
<th>Feature 2</th>
<th>Feature 3</th>
</tr>
</thead>
<tbody>
<tr>
<td>0-10</td>
<td>45</td>
<td>0</td>
<td>770</td>
</tr>
<tr>
<td>10-20</td>
<td>0</td>
<td>64</td>
<td>0</td>
</tr>
<tr>
<td>20-30</td>
<td>40</td>
<td>48</td>
<td>1</td>
</tr>
<tr>
<td>30-40</td>
<td>530</td>
<td>0</td>
<td>179</td>
</tr>
<tr>
<td>40-50</td>
<td>0</td>
<td>1212</td>
<td>0</td>
</tr>
<tr>
<td>50-60</td>
<td>1191</td>
<td>0</td>
<td>928</td>
</tr>
<tr>
<td>60-70</td>
<td>1293</td>
<td>0</td>
<td>1174</td>
</tr>
<tr>
<td>70-80</td>
<td>1446</td>
<td>0</td>
<td>1454</td>
</tr>
<tr>
<td>80-90</td>
<td>1387</td>
<td>0</td>
<td>1414</td>
</tr>
<tr>
<td>90-100</td>
<td>1235</td>
<td>0</td>
<td>1437</td>
</tr>
</tbody>
</table>

Table 5
Number of Boiling and Non-boiling Decisions for File 12

<table>
<thead>
<tr>
<th>Time in Sec.</th>
<th>Feature 1</th>
<th>Feature 2</th>
<th>Feature 3</th>
</tr>
</thead>
<tbody>
<tr>
<td>0-10</td>
<td>0</td>
<td>44</td>
<td>0</td>
</tr>
<tr>
<td>10-20</td>
<td>0</td>
<td>49</td>
<td>0</td>
</tr>
<tr>
<td>20-30</td>
<td>3</td>
<td>37</td>
<td>0</td>
</tr>
<tr>
<td>30-40</td>
<td>4</td>
<td>19</td>
<td>0</td>
</tr>
<tr>
<td>40-50</td>
<td>8</td>
<td>10</td>
<td>0</td>
</tr>
<tr>
<td>50-60</td>
<td>24</td>
<td>5</td>
<td>9</td>
</tr>
<tr>
<td>60-70</td>
<td>65</td>
<td>1</td>
<td>27</td>
</tr>
<tr>
<td>70-80</td>
<td>101</td>
<td>0</td>
<td>48</td>
</tr>
<tr>
<td>80-90</td>
<td>104</td>
<td>0</td>
<td>39</td>
</tr>
<tr>
<td>90-100</td>
<td>90</td>
<td>0</td>
<td>42</td>
</tr>
</tbody>
</table>
Here, it may be noted that at the time of boiling initiation both boiling and non-boiling decisions are present, though less compared to that in the fully developed boiling region.

Fig. 2 gives boiling and non-boiling decision as function of time for feature 1 and file 1.

**Fig. 2. Boiling and Non-Boiling Decisions as a Function of Time**

It should be noted that up to 26.8 sec, there is no boiling decision. After 28.3 sec, all decisions are boiling decisions. In between these two times both boiling and non-boiling decisions are seen. Also the time taken per decision is large around the time of onset of boiling. Fig. 3 presents similar data for file 12.

**Fig. 3. Boiling and Non-Boiling Decisions as a Function of Time**

Here we see that there are two boiling decisions before 27.8 sec. and are perhaps spurious. Also only after 63 sec. we get all decisions as boiling.

5. CONCLUSIONS

From the above study we conclude the following:

1. The Random Walk technique is found to be good in predicting the onset of boiling, especially, for file 1 where signal to noise ratio is good.

2. The technique takes very small time to take a decision both under normal and boiling conditions. But around the onset of boiling, the time taken is more.

3. The observed time taken for taking decision in background and boiling region agree well with the theoretical times.

4. Around the time of onset of boiling, we find that our procedure leads to many wrong decisions, more so for file 12 as compared to file 1.
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PARAMETERIZATION OF IN-CORE PWR SIGNALS FOR USE WITH PATTERN RECOGNITION TECHNIQUES

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Abstract - A parameterization of in-core and ex-vessel signals is presented. This set of parameters can be used to quantify the state of the vibrating core internal components in an operating commercial PWR. Coordinated in-core and ex-vessel noise measurements in an operating PWR are the basis upon which the parameters are estimated.

KEYWORDS
In-core, Ex-vessel, Partial Spectrum, Monitoring

INTRODUCTION
To quantify the vibration of the core internals, as transmitted via the fluctuations of the neutron field, in an operating PWR a set of parameters can be developed that allows the comparison (either over time or between plants) of the state of vibrating internals. These parameters, which have physical significance, make it possible to reconstruct some of the time series descriptors. The first section in this paper describes such a method and produces a list of parameters based upon the noise measurements detailed earlier (Dailey and Albrecht 1984).

METHOD
In a review of earlier papers by several authors models were suggested for the various descriptors of the in-core and ex-vessel signals (Kosaly 1979). These were based upon idealized physical descriptions of the various components. Using these models the relevant features of the APSD's can be quantified using a few parameters.

Previous studies have stated that the in-core neutron signal is proportional to the magnitude of the vibration of various core structures. The information is transmitted to the detectors via the neutron field. It has been suggested that the in-core signals are sensitive to the following contributions in the 2 to 40 Hz frequency range:

• Lateral motions of the fuel assemblies,
• Lateral motion of the CSB,
• Noise,
• Localized assembly specific information.

The model is a superposition of these components. The portion of the spectra attributed to the motion of core internal structures is developed from equations of motion for the fuel assembly skeleton and the CSB. The localized assembly specific information is obtained from the partial spectrum of the conditioned in-core signals.

PARAMETERIZATION
The fuel assemblies are constructed of a long slender skeleton from which the fuel pins are hung using spacer grids. These grids are dimpled and act as a spring support for the fuel pins. The skeleton is in compression due to the weight of the upper core plate, and is positioned on alignment pins on both the top and bottom.

The assembly is modeled as a long slender bar, of length L, for which shear and rotational inertia effects can be neglected. The equation used to describe the lateral motion of such a structure is,
\[
\frac{\partial^2}{\partial x^2} \left( EI(x) \frac{\partial^2 y(x, t)}{\partial x^2} \right) + \frac{\partial}{\partial x} \left( P \frac{\partial y(x, t)}{\partial x} \right) + k(x) y(x, t) \\
+ m(x) \frac{\partial^2 y(x, t)}{\partial t^2} + C(x) \frac{\partial y}{\partial t} = f(x, t)
\]

where,

- EI - is the stiffness,
- P - is the constant axial preload,
- m - is the mass per unit length,
- k - is the spring constant associated with the spacers,
- C - is the viscous damping coefficient,
- f(x, t) - is the external lateral force.

And the solution form can be written as,

\[
y = \sum_{n=1}^{\infty} q_n(t) z_n(x).
\]

The spectrum resulting from the forced motion of this structure can be written,

\[
S_{ii}(\omega) = |H_d|^2 |H_m(\omega)|^2 S_{PP}(\omega),
\]

where,

- \( H_d \) - is the detector transfer function,
- \( H_m \) - is the fuel assembly transfer function,
- \( S_{PP} = F(\omega)^* F(\omega) \),
- \( F(x, \omega) = \frac{1}{z_i} \int_{-\infty}^{\infty} e^{-i\omega t} \left( \int_0^L z_i f(x, t) \, dx \right) \, dt. \)

With the assumption that the forcing function has little character in the frequency range of the first bending mode, and that the transfer function of the detector is approximately constant in this frequency range, then the spectrum of an in-core signal can be approximated by a three parameter analog,

\[
APSD - S_{ii}(\omega) \approx \frac{A_2}{(A_1 - \omega^2)^2 + \omega^2 A_3}
\]

where,

\[
A_1 = \frac{k}{m} + \frac{1}{m} \left( \frac{EI}{z_i} \frac{\partial^2 z_i}{\partial x^2} + \frac{P}{z_i} \frac{\partial^2 z_i}{\partial x^2} \right)
\]

\[
A_2 = \left( \frac{1}{m} \right)^2 |H_d|^2
\]

\[
A_3 = \left( \frac{c}{m} \right)^2.
\]

The second part of the in-core signal is the CSB motion contribution. The in-core ex-vessel coherence is largest in the range of the CSB motion, larger than the coherence at the first bending mode for the peripheral assemblies. This implies that the in-core signals are sensitive to the flux fluctuation caused by the CSB motion.

Following Bauernfeind (1977) as well as Wach and Sunder (1977) the system description for the "beam" motion of the CSB in the frequency domain is written
Parameterization of in-core PWR signals

\[
H(\omega) = \frac{C_2}{-\omega^2 C_1 + \omega C_3}.
\]

(5)

The APSD of the ex-vessel signal is then written

\[
S_{YY}(\omega) = |H(\omega)|^2 S_{\phi \phi}(\omega),
\]

(6)

where,

- \( S_{\phi \phi} \) is the forcing function for CSB motion,
- \( H(\omega) \) is the transfer function for the CSB.

As with previous authors the forcing function will be assumed to have little character in the relevant frequency range (Kosay 1979). This allows the function

\[
S_{YY}(\omega) \approx \frac{C_2}{(C_1 - \omega^2)^2 + \omega^2 C_3}
\]

(7)

to be representative of the shape of the ex-vessel APSD in the CSB beam motion frequency range. The values of the coefficients \( C_1, C_2, \) and \( C_3 \) are determined by fitting equation (7) to the nearest ex-vessel APSD. An example of this fit is shown in figure 1.

The third identified contribution is the presence of noise in the signal. The noise component is included as a white noise background. This is modeled as a constant in the frequency domain.

The fourth contribution to the in-core signal spectrum is the information localized to the individual assembly. Partial spectral techniques as described in an earlier paper (Dailey and Albrecht 1984) are used to produce a conditioned spectrum that contains no information correlated with signals from surrounding assemblies. This is then representative of the information that is particular to a selected assembly. A comparison of the original spectrum and the conditioned partial spectrum is shown in figure 2.

Thus the APSD of an in-core signal is written,

\[
S_{ii}(\omega) \approx \frac{A_{2i}}{(A_{1i} - \omega^2)^2 + \omega^2 A_{3i}} + \frac{A_{4i}}{(C_1 - \omega^2)^2 + \omega^2 C_3} + A_{4i} + S_{ii+1}(\omega)
\]

(8)

where, \( S_{ii+1}(\omega) \) is the partial spectrum.

This identifies six parameters to reconstruct the in-core signal APSD; five constants and a frequency dependent partial spectral function. The parameters are calculated using a weighted nonlinear fit of the three physical analogs and the partial spectra to the observed in-core spectra (Marquardt 1963). A comparison of the spectrum of an interior assembly and the model is shown in figure 3.

The correspondence between the original and the model is nearly exact. Since this separates the large scale structures from the localized information in the in-core spectra it is a good starting place for identifying and quantifying the state of the reactor internals.

**MONITORING**

In addition to the in-core spectrum parameters, several other descriptive constants have been identified. For example, the maximum RMS displacement and CSB dominant angle of motion are evaluated (Dailey and Albrecht 1984). With this set of parameters and the ex-vessel APSDs the state of the reactor internals, as they are sensed by the neutron field, can be quantified. The use of this selection of parameters examines both the large core structures, through the ex-vessel signals, and fuel assemblies via the in-core signals.

The parameter vector for a fuel assembly is
\[ X_{ij} = \begin{bmatrix} a_{1ij} \\ \vdots \\ a_{6ij} \\ \omega_{k,j} \\ \xi_{k,j} \\ \xi_{(k-1)} \end{bmatrix} \]  

where,

- \( a_{k,i,j} \) - is the \( k \)th parameter of the \( i \)th detector in the \( j \)th measurement,
- \( \omega_{k,j} \) - is the natural frequency of the \( k \)th detector in the \( j \)th measurement,
- \( \xi_{k,j} \) - is the damping of the \( k \)th detector in the \( j \)th measurement.

A vector to characterize the large core internal structures is

\[ Y_j = \begin{bmatrix} \epsilon_j \\ \beta_j \\ C_{3j} \\ C_{2j} \\ S_{AA} \\ \cdots \\ S_{DD} \end{bmatrix} \]  

where,

- \( \epsilon_j \) - is the corrected RMS CSB displacement,
- \( \beta_j \) - is the dominant angle of CSB motion,
- \( S_{AA}(\omega) \ldots S_{DD}(\omega) \) - are the ex-vessel APSD's.

Each of these vectors has six members with some being frequency dependent. A set of vectors indicating the state of the reactor might be constructed as

\[ \mathbf{M} = \begin{bmatrix} X_{1j} \\ X_{2j} \\ \vdots \\ X_{nj} \\ Y_j \end{bmatrix} \]  

where \( n \) is the number of fuel assemblies chosen to monitor.

A representative value for each of the elements in this construct can be obtained from a series of measurements. The elements are constructed from the mean value for the parameters over a set of observations during a learning period. For example,

\[ a_i^J = \frac{1}{N} \sum_{j=1}^{N} a_{i,j}^J. \]  

Further, the variability of the measurements can be addressed by creating a similar construct for the variance of each of the elements,

\[ \mathbf{Q} = \begin{pmatrix} 1 & 0 & \cdots & 0 \\ 0 & 1 & \cdots & 0 \\ \vdots & \vdots & \ddots & \vdots \\ 0 & 0 & \cdots & 1 \end{pmatrix} \frac{1}{\sigma^2_{ij} J} \]  

\[ (\sigma_{ij}^J)^2 = \frac{1}{N} \sum_{j=1}^{N} (a_{i,j}^J)^2 - (a_i^J)^2. \]
These two vector sets can then be used to monitor the state of the reactor. Significant changes in the fuel structure and vibration will affect one or more of these variables causing it to deviate from the established mean value. The scale of the deviation is based upon the variability of the measurements. If this deviation remains within some acceptable level, the state can be viewed as unchanged. If the deviation is beyond some threshold then a change in some internal component is indicated.

To evaluate the present state of the reactor internals, as sensed by the neutron field, the state vector set $\mathbf{M}$ is constructed. A squared deviation from the existing state is established by subtracting and squaring, and then normalizing by dividing the individual elements in the $D$ construct by their counterpart in the variance matrix $\mathbf{C}$.

$$D = (\mathbf{X} - \mathbf{M}) \mathbf{C} (\mathbf{X} - \mathbf{M}).$$ (14)

For example, a value greater than unity for any member means that element is more than one standard deviation from the established state. If the variability of the measurements is assumed to be normally distributed then a confidence level can be selected as the boundary used to indicate a change of state.

This methodology provides a means of establishing a baseline state, evaluating the deviation of the present state from the established baseline, and suggests a means to set a threshold if desired.

REFERENCES


Figure 1. Exvessel Spectrum and Three Parameter Model of CSB Motion.

Figure 2. Comparison of In-core Spectrum and the Conditioned Partial Spectrum.
Figure 3. Six Parameter model and In-core Spectrum.
CHARACTERIZATION OF CHAOTIC, NONSTATIONARY TIME SERIES AND ITS APPLICATION TO EARLY DETECTION OF ANOMALIES IN NUCLEAR REACTORS

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Abstract - A statistical theory of characterization of chaotic, nonstationary time series is presented. The theory is based on the identification method by the stochastic non-linear models. Time-domain analysis of stochastic process of jump events and of the bispectrum are the key methods to the present identification theory. It is emphasized that (i) the methods are sensitive to extract the non-linear nature of the systems; (ii) the generalized fluctuation-dissipation theorem for the non-linear Langevin equation becomes important for the precise identification of stochastic systems; and (iii) the methods are useful to detect anomalies in nuclear reactors.

1. INTRODUCTION

After the accident at the 4th unit of Chernowyl nuclear power station, the accuracy of the integral analysis for the plant behaviour on the sequence of events during the accident has been growing interests for many researchers. When an accident (or anomaly) takes place in a nuclear reactor, some non-linear modes may appear abruptly or some non-linear interactions may grow up slowly during the development of the accident (or anomaly). The temporal evolution of a relevant physical quantity may have generally nonstationary and chaotic nature (cf. Williams, 1974; Ash, 1979; Saito, 1983; Wakabayashi and Nariai, 1986).

To study nonstationary stochastic processes of sequences such as seismic waves, the six sophisticated spectra (Evolutionary spectrum, Priestly (1965); Instantaneous spectrum, Page (1952); Generalized spectrum, Bendat and Piersol (1966); Physical spectrum, Mark (1970); Multifilter spectrum, Kameda (1975); Developing spectrum, Hino (1975)) have been proposed (see Hino, 1977 and references cited therein). An appropriate general methodology does not exist for all types of nonstationary random data. It follows that various special techniques that apply a limited class of nonstationary data must be studied to detect anomalies.

To exhibit theoretical methods for characterizing a class of nonstationary sequences and to demonstrate their feasibility for early anomaly detection are the main purposes of the present paper. Firstly, we show a method of time-domain analysis for quasi-nonstationary sequences which are characterized by appearance of hump event or bursting event. The key ideas to identify the sequence with hump event are as follows: (i) The experimentally obtained probability density function (pdf) is fitted by the theoretical pdf due to the Smoluchowski approximation; (ii) To determine the rate of appearance of hump event and infer the damping constant and the strength of noise. However, one encounters frequently more complex (multiple-components) or more noisy sequence in turbulence which is a paradigm of real anomalous nonstationary sequences. Therefore, in this paper we develop the above time-domain analysis to apply the systems which can be identified by the parametric non-linear stochastic systems. The system identification can be possible by virtue of a generalized fluctuation-dissipation theorem for the systems described by the stochastic nonlinear differential equations.

Secondly, an anomaly detection method with use of the bispectrum is proposed. To demonstrate the feasibility of the bispectral analysis for the early detection of anomalies, we must show the sensitivity of the bispectral analysis. We exhibit its high sensitivity by many examples: e.g., (i) Sensitivity of the bispectral pattern for small change of effective potential by using the non-linear stochastic models; (ii) Sensitivity for a complex non-linear signal that the contribution ratio of a non-linear component in the total power is fairly small.

The paper is organized as follows: Section 2 introduces the non-linear characteristics of the stochastic processes in nuclear reactors. Then the necessities of a time-domain analysis and the higher order correlation functions are explained. Section 3 exemplifies the theory of identification stated in section 2 for the stationary sequences which can be regarded as quasi-nonstationary sequence for a short duration of time. Here we demonstrate the method of analysis and the availability of the generalized fluctuation-dissipation theorem. Section 4 explains the method of the bispectral analysis and its feasibility of early warnings of anomalies in nuclear reactors. The final section is devoted to the concluding remarks.
2. THEORY OF CHARACTERIZATION OF CHAOTIC, NONSTATIONARY SEQUENCE

2.1 Non-linear Phenomena in Nuclear Reactors

Non-linear nature appears in the random processes in nuclear reactors. The relatively simple examples are classified into the four groups and listed as follows.

A. Anomalous growth of fluctuation
   (a) Anomalous power fluctuation in the BORAX reactor and the high power BWRs (Ackasu, 1965; Williams, 1974; Konno, 1985; Enomoto et al., 1985; Leuba, Cacuci and Perez, 1986).
   (b) Anomalous fluctuation at the transient state (the starting up of a reactor) (Quabili, 1981; Konno, 1984, 1985).

B. Pulsation, Hump, Jump Phenomena
   (a) Hump Phenomena arose due to the anomalous core-barrel motion (Friy, Kryter and Robinson, 1974; Thie, 1979; Konno, 1986).
   (b) Hump phenomena due to thermal convection systems (Ahlers and Walden, 1980).
   (c) Pulsation and shock due to the motion of two-phase flow (Wijngaarden, 1972; Fukano and Osaka, 1985)
   (d) Transmitted neutron flux fluctuation dependent upon the flow patterns in two-phase flow (Ablech et al, 1982)

C. Bursting Phenomena
   (a) Turbulent sequence of fluid velocity (or pressure) (Frisch, 1985)
   (b) Acoustic Emission from bubbles (Dentico et al, 1982)
   (c) Borselle Pressure Noise (Turkcan, 1985)
   (d) Sound due to loose parts impact and its propagation (Izumi et al., 1984; Mayo, 1985)

D. Other Non-Linear Phenomena
   (a) Non-Gaussian temperature fluctuation due to the blockage of a coolant channel (Turkcan, 1977)
   (b) Non-Gaussian pressure fluctuation in a pipe in liquid-vapor two phase flow (C. Matsui, 1984)
   (c) Resonant seesaw phenomena in the Haldane reactor due to thermo-hydraulic interactions (Oguma, 1980)

The governing equations for describing these systems are complex ones which seem to be quite different physically and mathematically from each other. However, the various similarities also exist from the viewpoint of the nonequilibrium statistical mechanics.

2.2 Feature of Chaotic, Non-stationary Sequence

The general features of the nonlinear phenomena which can be classified as chaotic and non-stationary nature are summarized as follows:
(1) the variance, the higher-order moments deviate from the Gaussian nature (\( \mu_3=0, \mu_4=3\sigma^4 \));
(2) the probability density function with non-Gaussian character;
(3) the multiplicity of peaks in the power spectrum of fluctuations and their mutual correlation;
(4) the long time correlation (the long time memory or the long time tail) arises;
(5) Pulsations, humps (Jump phenomena) and shock appear [the pattern forming mechanisms works strongly due to non-linearity under the non-equilibrium situations];
(6) Anomalous and/or turbulent diffusion take place.

From the viewpoint of anomaly detection, the complex models which require elaborate computations are not recommended. One may agree with the following principles of detection of anomaly:
(1) Simple model for identification is preferable;
(2) It is also useful for the pattern recognition of anomalies.

2.3 Necessity of Time-Domain Analysis and Higher Order Correlations

The chaotic non-linear sequences possess frequently long time correlation as remarked in the previous subsection. To analyze these sequences precisely, it is necessary to take a long-time record. In addition, the analysis of sequences in the nonstationary cases must be accomplished by calculation on single record (Bendat and Piersol, 1986; Chapter 12). To construct the efficient method to detect the early warning of anomalies, one must discard the rigorous equivalence of the time average to the ensemble average.

The most fundamental method for detect the non-linear nature is to observe the temporal development of the variance (Otsuka and Saito, 1981), the skewness \( S(t) = \mu_3 / \sigma^4 \) (cf. Turkcan, 1977), the flatness \( F(t) = \mu_4 / \sigma^4 \) and the intermittency factor \( I_2(t) \) (Schubauer and Klebanoff, 1956; Konno, 1985). However, these are the integrated quantities of fluctuating state variables. To know more precisely the behaviour, the nature and the status of systems, one must use the higher order statistical quantities such as the frequency spectra.
For the qualitative understandings of the nonstationary situations, one can use the method of linear analysis in a time domain. Data are divided into small pieces with a constant time interval $T$. Then (i) the time development of the variance $\sigma(nT)$ ($n$ integer), the PSD $P(\omega;nT)$ (cf. Turcak and Kitamura, 1985), the noise contribution ratio $RNC(\omega;nT)$ and the partial coherence $PC(\omega;nT)$ are calculated by the AR (ARMA) method (the main interests are the PSD and the variance); (ii) multiple-variables AR method is used to obtain the time development of the partial coherence (cf. Oguma and Turcak, 1985), the cross power spectral density $CPSD(\omega;nT)$, the noise contribution ratios (the main interests are interpretations of the interactions of physical quantities). Continual observation of these quantities may give useful information of early warnings of anomalies.

However, the PSD is not always powerful to understand the features of the relevant temporal sequence (the inner status of a relevant system). Namely, for some class of complex noises the linear analysis can not give us available information to distinguish the situations of the system. There are great many types of sequences which give rise to the same power spectral density. The simplest examples that the different three processes give the same PSD profile $P(\omega) = 1/(\omega^2 + \lambda^2)$: (a) the linear Markovian process described by the Langevin equation $\ddot{x} + \lambda \dot{x} + f(t)$ driven by the Gaussian white noise; (b) the occurrence pulse or hump event obeying the Poisson process, viz. $P(n,t) = (-\lambda t)^n \exp(-\lambda t)/n!$; (c) The chaotic sequences obtained from the non-linear transformation $x_{n+1} = f(x_n)$ (a Bernoulli transformation and a tent transformation) (cf. Mori, So and Ose, 1981).

The method of non-linear analysis in a time domain is not only necessary but exhibits distinctly the differences between them. There are at least two methods available to distinguish them. The first one is to study the stochastic process of jump events or to study the recurrence time of the relevant sequence. The second one is to study the higher order correlation functions. The chaotic sequences described in the above category (c) give the exactly the same PSD. Namely, they are not distinguished via the second order correlation function $C_r(\tau) = \langle X(t+\tau)X(t) \rangle$. However, the recurrence time distributions with various counting thresholds of amplitude of sequences give information to distinguish the two data. Similarly, the higher order correlation functions $C_{2r}(\tau_1, \tau_2, \ldots) = \langle X(t+\tau_1)X(t+\tau_2) \ldots X(t) \rangle$ or their Fourier transforms become different with each other. The microscopic processes reflect sensitively to the higher order correlation functions. The argument described above may be also applied to the nonstationary cases though the situations become generally complex.

### 3. TIME DOMAIN ANALYSIS OF JUMP EVENT AND BURSTING EVENT

#### 3.1 Stochastic Theory of Jump and Bursting Phenomena

Consider the theory of jump and bursting phenomena based on the stochastic non-linear differential equations (SDE);

$$
\dddot{x} + (a + b(x) + \xi(t)) \ddot{x} + \frac{\partial}{\partial x} V(x) = f(t).
$$

(3.1)

where $V(x)$ is the non-linear effective potential, $f(t)$ and $\xi(t)$ are the additive and the parametric noise, respectively. Although this is a simple model, various features of a number of systems with infinitely many degrees of freedom can be simulated.

Case (I) $a = k$ ($k > 0$), $b(x) = 0$ and $\xi(t) = 0$;

$$
\dddot{x} + k \ddot{x} + \frac{\partial}{\partial x} V(x) = f(t).
$$

(3.2)

When the hump phenomena is observed in a time series, and the pdf has a non-Gaussian nature, the system may be identified by the non-linear model. In this case, the stochastic process of hump events are the key to determine the system parameters and to understand the nature of non-linear phenomena. Procedure of the identification is illustrated in Figure 1. The expressions of the rate of hump event during the time interval $T$ derived from the generalized Polya process which covers various simple stochastic processes (Konno, 1986) are summarized in Table 1.

The study of the stochastic process plays an important role to determine $k$ and $D$ (cf. (3.4)).

Case (II) $a = \alpha$ ($\alpha > 0$), $b(x) \neq 0$ and $\xi(t) \neq 0$;

$$
\dddot{x} + \alpha \ddot{x} + \frac{b(x) - \xi(t)}{\partial x} \ddot{x} + \frac{\partial}{\partial x} V(x) = f(t).
$$

(3.3)

The occurrences of (violent) bursting can be explained by the model. Procedure of the identification is illustrated in Fig. 2. In this case, the study of the stochastic process of the jump event do not become useful as was achieved in Case (I). The theoretical modeling of the generalized narrow band process becomes the key to identify the relevant sequences.
3.2 Generalized Fluctuation-Dissipation Theorem

A generalized fluctuation-dissipation theorem (FDT) for the non-linear Langevin equation (3.1) can be constructed as shown below.

Case (I) : The Fokker-Planck equation for the Langevin equation (3.2) within the Smoluchowski approximation gives the stationary solution in the following form:

\[ P_s(X) = N_0 \exp\left( -\frac{2kV(X)}{D} \right) \tag{3.4} \]

where \( N_0 \) is the normalization constant, \( V(X) \) is the effective potential, \( D \) is the diffusion constant, and the \( k \) being the damping constant. According to the numerical simulation (Konno, 1986), the theoretical probability density \( P_s(X) \) stands for smaller damping constant \( (k>0.01) \) if a long-time record of simulated sequence is sampled.

We notice that the probability density \( P_s(X) \) takes exactly the same form as the non-linear diffusion process. For the one dimensional diffusion process, a generalized fluctuation-dissipation theorem has been proposed (Okabe, 1985). The theorem relates among the variance \( \sigma^2 \), the diffusion constant \( D \) and the generalized friction \( C_{\beta,\gamma} \) in terms of the probability density \( P_s(X) \) and the power spectral density in the following form:

\[ \begin{align*}
D &= D/2k^2 \quad \beta_{\beta,\gamma} \tag{3.5a} \\
\sigma^2 &= \int_{-\infty}^{\infty} x^2 P(X) \, dx = \frac{1}{2\pi} \int_{-\infty}^{\infty} P(\omega) \, d\omega \tag{3.5b} \\
\beta_{\beta,\gamma} &= \int \left| \beta + \gamma(\omega) - i\omega \right|^2 \, d\omega \tag{3.5c} \\
&= \frac{1}{k} \int \frac{3}{\partial X} V(X) \frac{\partial}{\partial X} P(X) \, dx / \sigma^2. \tag{3.5d}
\end{align*} \]

The parameter \( \beta \) and the function \( \gamma(t) \) in eq. (3.5c) are defined in the formal linear Langevin equation corresponding to the Smoluchowski approximation of the non-linear Langevin equation (3.2);

\[ \ddot{X}(t) + \beta X(t) - \int_{t}^{\infty} X(t+u) \gamma(u) \, du + \alpha \dot{W}(t), \tag{3.6} \]

where \( W(t) \) is the Wiener process which strength is unity, viz., \( \langle \dot{W}(t)\dot{W}(t') \rangle = \delta(t-t') \). The generalized FDT (3.5) can be used not only to identify the system parameter without studying the stochastic process of jump event but to check the consistency of the identified system parameters as shown in Fig.1.

Similar argument can be developed for the non-linear systems with a parametric noise:

\[ \ddot{X} + k \dot{X} + \frac{3}{2} V(X) = \xi(t). \tag{3.7} \]

The details are not described here. The generalization to a two-dimensional non-linear system will be reported in near future (Konno, 1987).

Case (II) : For the systems described by the equation (3.3), the method of analysis does not become simple as described above. For simplicity, let us consider the special case;

\[ \beta(X)=8X^2, \quad V(X)=\omega^2 X^2/2 \quad \text{and} \quad \xi(t)=0. \tag{3.8} \]

By assuming

\[ X(t)=A(t) \sin(\omega_0 t + \Phi(t)), \tag{3.9} \]

where \( A(t) \) and \( \Phi(t) \) are the slowly varying functions, we can obtain the set of Langevin equations for the amplitude and the phase in the following form;

\[ \frac{d}{dt} A(t) = \frac{1}{2} \alpha A - \frac{1}{8} \beta A^3 + \frac{1}{2} \xi(t), \tag{3.10} \]

and

\[ \frac{d}{dt} \Phi(t) = \frac{1}{2} \sin(2\omega_0 t + 2\Phi(t)) \xi(t). \tag{3.11} \]

Since (i) the amplitude and the phase are decoupled with each other, and (ii) the obtained equation (3.10) takes the same form as the Smoluchowski approximation of the parametric equation (3.7), the FDT can be also applied for the motion of the amplitude.
Fig. 1 Identification scheme of the effective potential $V(x)$ for the non-linear model:

$$\ddot{x} + k \dot{x} + \frac{\partial}{\partial x} V(x) = F(t), \quad \langle F(t) F(0) \rangle = D \delta(t).$$

<table>
<thead>
<tr>
<th>Table 1 Stochastic Process of Jump Event</th>
</tr>
</thead>
<tbody>
<tr>
<td>$\frac{d}{dt} P(n,T) = \lambda_{n-1}(t) P(n-1, t-T_0) e^{-\lambda T_0 n - \lambda_0} P(n, t-T_0) e^{-\lambda T_0}$</td>
</tr>
<tr>
<td>$\lambda_n(t) = \frac{1 + \alpha}{1 + \alpha t}$</td>
</tr>
</tbody>
</table>

(i) $T_0 \rightarrow 0$

$P(n,T) = \left(1 + (1+2\alpha)(n+1)\right) \ldots \left(1+\alpha T - \lambda/\alpha T\right)^{-\lambda/\alpha T} \left[1 - (1+\alpha T - \lambda/\alpha T)^{-\lambda/\alpha T}\right]$^{n-1/n!}$

(ii) $\alpha \rightarrow 0$

$P(n,T) = \sum_{m=n}^{\infty} \lambda \frac{(m-n)}{m!} e^{-\lambda T_0} (T-T_0)^m$

(iii) $P(n,T)$

$\beta = 1 \quad$ Polya Process

$\beta = \lambda, \alpha = 1 \quad$ Yule Process; $P(n,t) = \exp(-\lambda t)[1 - \exp(-\lambda t)]^{n-1}$

$T_0, \alpha \rightarrow 0 \quad$ Poisson Process; $P(n,T) = (\lambda T)^n \exp(-\lambda T)/n!$
3.3 Applications
(a) Hump Phenomenon

The model gives a qualitative explanation of the hump phenomenon of neutrons observed in the Palisades which appears due to the anomalous core-barrel motion (Fry et al., 1974). The model explains qualitatively not only the stochastic process of hump events but the profile of the pdf although we neglects the information conversion process (the neutron transport process to the detector) (Konno,1986). The fluctuation signal of the neutron in the Palisades is traced in Fig.3 and the numerically simulated signal is illustrated in Fig.4. Since the rate of the appearance of the hump is not frequent, the frequency-domain analysis with use of the short record of sequence is not adequate to infer the situation of the system.

The Fourier transform of the higher order correlation functions such as the bispectrum may be utilized for the detailed analysis of the anomalous vibration (cf. Section 4).

Fig. 2 Identification scheme of the bursting phenomena \( \mu_4 = 0 \) for the generalized narrow band process; \( x(t) = A(\xi) \sin(\omega_0 t + \phi(t)) \).

Theoretical Model of Bursting Phenomena:
Stochastic modeling of the amplitude \( A(t) \) and the phase \( \phi(t) \)

Theoretical estimates of \( B(\omega_1, \omega_2), P(\omega) \), \( F \) and \( S \)

The generalized Fluctuation-Dissipation theorem
\[ \mathcal{D} = \sigma^2 \mathcal{C}_{\beta, \gamma} \]
(b) Bursting phenomenon

As an example of the bursting phenomena, we show in Fig. 5 the pressure fluctuation signal observed in the Borselle reactor. The Borselle pressure noise is characterized by the multiplicity of the peaks and the large flatness (F=3.5) than the Gaussian distribution (F=3). When the bursting sequence is analyzed by setting the threshold of amplitude, one obtains the jump number distribution P(n,T) with a nonstationary nature which is identified by the generalized Polya process (Konno, 1986). The large variance σ² calculated from P(n,T) is due to the slow modulation of the amplitude. For the bursting phenomena the study of the stochastic process is not efficient so far due to the lack of the one-to-one correspondence with the theoretical models (3.3) and the parameters of the generalized stochastic process.

The bursting phenomena is ubiquitous in the thermo-hydraulic systems in nuclear reactors. Qualitatively, the occurrence of the bursting can be understood by the narrow band process or the band limited white noise. Then the important points are (a) whether there are multiple peaks in the power spectral density of the relevant fluctuation ?; (b) whether their peaks are mutually correlated ?; and (c) whether the flatness is large ?

To describe the narrow band process we introduce the amplitude A(t) and the phase φ(t) in the form: x(t)=A(t) sin( ω0t + φ(t)). The probability density function derived from the stochastic Van der Pol model: P (A)=A² exp(-BA²/D) (B > 1), we can explain the growth of the flatness. By introducing the stochastic model of a quasi-periodic oscillation of the phase, multiplicity of the peaks can be obtained (Konno et al., 1987). To analyze the bursting phenomena more precisely, one must calculate further the bispectrum as described in the next section.

![Fig. 5] Pressure fluctuation observed in the Borselle reactor.

4. BISPECTRAL ANALYSIS

4.1 Definition

The bispectrum is defined by (Hasselman, Munk and McDonald, 1963)

$$B(\omega_1, \omega_2) = \frac{1}{(2\pi)^2} \int \int \omega_1 \omega_2 < X(t)X(t+\tau_1)X(t+\tau_2) > \exp(-i\omega_1 \tau_1 - i\omega_2 \tau_2).$$

(4.1)

The biocherence and the biphase are frequently used to display the distortion of the bispectrum by the non-linearity. However, these quantities are sensitive to the numerical errors and/or the smoothing algorithm (the spectral windows). We therefore use the original bispectrum in the logarithmic scale. The bispectrum has the symmetry B(ω1, ω2)=B(ω2, ω1)=B(−ω1, ω2)=B(−ω1, −ω2). The bispectrum throughout the paper is therefore illustrated in the minimal triangle in the two dimensional space: (ω1, ω2)=(0,0), (fN/2,fN/2) and (0,fN) where fN=(1/2πc) denotes the Nyquist frequency. [# We will show log|B(ω1,ω2)| throughout the paper.]

4.2 Physical Meaning

The standard spectral quantity is the power spectral density P =< X[ω1] X[ω2] > (ω1, ω2=0). This expresses the interference between two modes. The bispectral density can be also expressed as

$$B = < X[ω_1] X[ω_2] X[ω_3] > (ω_1 + ω_2 + ω_3 =0).$$

(4.2)

This expresses the interference among the three waves. Similarly, trispectrum is defined as

$$T = < X[ω_1] X[ω_2] X[ω_3] X[ω_4] > (ω_1 + ω_2 + ω_3 + ω_4 =0).$$

(4.3)

It should be noted that σ² =∫ P(ω) dω, υ₁ =∫∫ B(ω₁,ω₂) dω₁ dω₂ and υ₄ =∫∫∫ T(ω₁,ω₂,ω₃) dω₁ dω₂ dω₃. The relation implies that: Consider the situation that the temporal development of a relevant variable at the normal state is characterized by υ₃=0. If in an anomalous state the system is characterized by the growth of the third order moment, the bispectral analysis may become useful to detect the anomaly. Similarly, when the anomalous state is characterized by υ₄=0 and the growth of the values of the flatness (i.e. the 4th order moment), it is expected that the peak intensity of the trispectrum increases as the flatness increases.
4.3 Applications
(a) Non-stationary sequence

Consider the sequence (Fig.6) where a linear fluctuation is abruptly interrupted by a non-linear fluctuation after the time indicated by the arrow. The linear and the non-linear fluctuation is described by the Langevin equation (3.2) with the potential $V(X)=A_4X^2/2$ ($A_4=400; k=1, D=20$) and $V(X)=A_2X^2/2+A_3X^3/3+A_4X^4/4$ ($A_2=400, A_3=100, A_4=800; k=1, D=20$), respectively. Since the damping constant $D$ is large, the clear distinction between the two states does not appear in the power spectral densities of the fluctuations as shown in Fig.7. As expected from the potential change seen in Fig.8, the peak intensity of the PSD in State 2 becomes smaller than that of State 1. However, the bispectrum in Fig. 9 shows clearly the change of the state due to the appearance of non-linearity.

Fig.6 Simulated nonstationary time series. The onset time of non-linear oscillation is indicated by the arrow →.

Fig. 7 PSDs before and after the onset of non-linear oscillation.

Fig. 8 Shapes of the Potentials in the Langevin equation (3.2) for the state 1 and 2.

Fig.9 Bispectral patterns before (State 1) and after (State 2) the onset of non-linearity.
(b) Bursting phenomenon

Figure 10 shows the bursting phenomena which is obtained from the equations described by coupled parametric oscillators (Equation (4.4)). When the noise $D=1$ is introduced, a random modulation of bursting takes place and new mode-mode couplings appear strongly as seen in State 2. The change of the feature of mode couplings does not understand from the power spectral density as shown in Fig.11.

\[ \begin{align*}
\ddot{x} + (-1 + \varepsilon(t)) \dot{x} + \omega_0^2 x &= 0 \\
\ddot{\xi} + (-1 + \xi^2) \dot{\xi} + \omega_1^2 \xi &= \epsilon(t) \\
\langle \epsilon(t) \epsilon(0) \rangle &= \delta(t) \\
\omega_0 &= 10 \\
\omega_1 &= 0.3
\end{align*} \] (4.4)

State 1 : $D=0$
State 2 : $D=1$

Fig. 11 The FSD of the coupled Van der Pol model (4.4). The envelope of the peaks obeys an inverse power law. There is no crucial difference between State 1 and State 2.

Fig. 12 Bispectral patterns of the non-linear oscillation for the periodic relaxation oscillation (State 1) and the modulatory oscillation with bursting (State 2).
5. CONCLUDING REMARKS

In this paper, we show a statistical mechanical theory of characterization of chaotic, nonstationary sequences in a time-domain by focusing our attention on the non-linear nature of random sequences. There are a number of situations that a non-linearity which is hidden when the system is close to the equilibrium state grows when anomalies, accidents, and malfunctions take place. The identification methods developed in this paper will be available for a limited class of non-linear random sequences in both the stationary and the nonstationary cases.

The non-linear modeling will be useful to predict and understand qualitatively (i) how the anomaly develops, (ii) how the growth of anomalous fluctuation becomes suppressed due to non-linearities, (iii) why the noble temporal patterns arise and so on. The more precise and practical studies accounting the ideas developed in the paper will become important in addition to the survey of the expedient and efficient signal-processing methods available for a large class of nonstationary sequences.

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REFERENCES

APPLICATION OF THE IDENTIFYING FUNCTIONS METHOD TO THE EX-CORE NEUTRON NOISE ANALYSIS OF PWRs

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Abstract - The method of the identifying functions is applied to various pressurized water reactors. As an interesting result of the application of this method to Borssele FWR, we observed the clear appearance of a shell mode core barrel vibration (SMCB) around 15 Hz which has not been reported during the SMORN-IV, although some calculations with VIBREAL and SPEC-4R indicated its possibility. A SMCB motion was found between 17 and 21 Hz in the three analyzed French FWRs.

1. INTRODUCTION

Recently, the phase criteria used to identify the peaks appearing in the auto and cross-power spectral densities (APSD and CPSD) of ex-core neutron detectors in FWRs (Bernard, 1977; Turkcan, 1982; Akerhielm, 1982) were expressed in a simple and useful way, through the introduction of different continuous functions of the frequency, which we called the "identifying functions" (Molina, 1987). Being a way to separate different components of the spectra according to its nature, this method may be extremely helpful for quick recognition of the patterns, especially in those situations when various peaks of different nature are very close in frequency. In this paper we present a resume of the method and its applications to Borssele and three French FWRs.

2. THE IDENTIFYING FUNCTIONS

The method consists in giving a quantitative measure of how close the six phases of the CPSDs between signals arising from four ex-core neutron detectors are to the ideal phase criterion determining one or another phenomenon. Two types of functions were defined (as many others could be defined): \( U_i(f) \) and \( V_i(f) \). The index \( i \) stands for the nature of the spectral component: \( R \) for reactivity effects; \( B \) for beam mode core barrel (BMCB) vibrations; \( SH \) for shell mode core barrel (SMCB) vibrations and \( BU \) for BMCB unidirectional vibrations. As an example we present here the definition of \( U_R, V_R, U_B \) and \( V_B \):

\[
U_R(f) = \frac{(2\pi - \phi)(4\pi - \Theta)}{8\pi^2} \quad ; \quad V_R(f) = -\log(\Theta + \phi) \quad (1)
\]

\[
U_B(f) = \phi/2\pi \quad ; \quad V_B(f) = -\log(2\pi - \phi) \quad (2)
\]

where \( \phi \) is the sum of two phases corresponding to the 180° opposite detectors, while \( \Theta \) is the sum of the four phases corresponding to the 90° adjacent detectors. The identity \( U_R + U_B + U_pm = 1 \), holds at any frequency.

In some sense the method of the identifying functions reminds the useful procedure introduced by Turkcan (Turkcan, 1982) analysing the direction of the BMCB motions as a continuous function of the frequency.

For the phases in the interval \((0, \pi)\), we found a simple approximation:
\[ \arctg \frac{I}{R} = \pi \left( \frac{\pi}{2} + \frac{I}{R} \right)^{-1/2} + C \] (3)

where R and I are the Real and Imaginary Parts of the CFPSD and C=0 for R>0 and C=\pi for R<0. Although this approximation practically does not distort the identifying functions it was not used in this paper.

3. APPLICATION TO BORSSELE PWR

Figs. 1 and 2 show the functions U_i(f) and V_i(f), respectively, for Borssele neutron noise. The noise source was the analog benchmark tape (FM) distributed among the participants of SMORN-III. The four signals used for the calculation of the identifying functions come from the ex-core detectors D62, D82, D52 and D72.

In Fig. 3 the Auto Power Spectral Density of one ex-core detector (D-62) is presented in arbitrary units. All curves were obtained in the frequency interval 0-40 Hz with a step \(f=0.125\) Hz.

As the Borssele spectrum was carefully studied in the previous SMORN, we shall not discuss it in detail and will limit ourselves to note some important features that can be seen from the identifying functions U_i and V_i.

- The reactivity peak at 9.2 Hz is well indicated by the functions U_l and V_l as one could expect.

- The core barrel vibrations in the interval 11-18 Hz (11.7, 12.7, 15.1, 16.0 and 17.6 Hz) summarized at the Borssele Physical Benchmark (Turckcan, 1984) are not completely reproduced by the functions U_u and/or U_l. Actually only two of them (12.7 and 16.0 Hz) appear in the functions U_u, V_u and U_s, V_s. The other three peaks appear in the identifying functions U_u and V_u showing its Shell Mode nature. It should be noticed that some possibility of shell mode core barrel vibration in this region was found by Yamada et al. (Yamada, 1984) applying both VIBREAL and SPEC-4R models. The small amplitudes of the obtained shell mode APSD compared with the beam mode APSD, made it difficult to arrive at a definite conclusion.

4. APPLICATION TO FRENCH PWRs

The method of the identifying functions was applied to three French PWRs denoted here as F-1 (955 MW); F-2 (937 MW) and F-3 (950 MW). The results are presented in Figs. 4-12 as follows: APSD, U_i and V_i for F1 PWR in Figs. 4-6. U_i, V_i and APSD for F2 PWR in Figs. 7-9. APSD, U_i and V_i for F3 PWR in Figs. 10-12. As for Borssele PWR the frequency interval is 0-40 Hz with a step \(f=0.125\) Hz.

All the functions U_i and V_i presented here were calculated using the signals of four neutron ex-core detectors of the upper plane of the core. The APSDs correspond to one of those detectors in each PWR. They are not normalized.

From Figs. 4-12 we can notice the following interesting features:

- The functions U_sh and V_sh in the three PWRs show neatly the existence of a SMCB vibration around 19 Hz. This peak, which is well observed in the APSDs, is the only one, according to U_sh and V_sh, observed in the F2 and F3 PWRs. In the F1 PWR another SMCB motion is indicated by the corresponding U_sh at 32 Hz.

- The functions U_s and V_s of the F2 and F3 PWRs are very similar; they show reactivity effects around 1, 10 and 15.5 Hz (at the last frequency, the peaks of U_s and V_s in both cases have a width of several Hz). In the F1 PWR these peaks are again indicated by the corresponding functions U_s and V_s although a wide frequency region of null phases appears for this PWR around 7 Hz and other higher frequencies.

- The functions U_s (and in less degree V_s) of the three considered French PWRs, show some similar behaviour at low frequencies. In F2 and F3 this function is close to unity in a wide frequency region from 2 to 10 Hz and around 11 Hz. In F1, the first region extends from 2 to 5 Hz and again U_s=1 from 11 to 15 Hz. Taking into account that in these PWRs U_sh=0 below 16 Hz, and therefore U_s+U_sh=1 it follows that the differences in the behaviour of U_s(f) of the three PWRs reflect the differences in the behaviour of U_s(f). In such situations, one could expect a mean value of U_s and U_sh of 1/2 except around the frequencies where a real phenomenon is taking place (U_sh=1 or U_s=1).
Fig. 1 The identifying functions $U_k(f)$ for Borssele PWR neutron noise.

Fig. 2 The identifying functions $V_k(f)$ for Borssele PWR neutron noise.

Fig. 3 The APSD of the signal of an ex-core detector in arbitrary units (Borssele PWR).

Fig. 4 The APSD of the signal of an ex-core detector in arbitrary units (French F1 PWR).
5. DISCUSSION

It is interesting to remark the similar behaviour of the functions \( V_{sh}(f) \) for the three considered French reactors, indicating the existence of a SMCB motion around 19 Hz. In the case of the Borssele PWR, \( V_{sh} \) and \( U_{sh} \) show that three peaks in the complex structure of the APSD between 11 and 17 Hz, may be considered as SMCB vibrations. The functions \( U_{a} \) (and \( V_{a} \)) have also a close behaviour for the three French PWRs, showing reactivity effects around 15-16 Hz and nearly 10 and 1 Hz. In Borssele PWR, this functions caught only the well known peak at 9.2 Hz. Finally the behaviour of the functions \( U_{a} \) of the four PWRs analyzed here have also many points in common.

The fact that the identifying functions are calculated using only the phases, and not the amplitudes of the APSDs, raise the question of their interpretation at frequencies where the random component of the noise predominates and may create spurious peaks. On the other hand, this same fact, probably make them very sensitive to different sources of noise, even when the amplitudes are small. An indication that a genuine effect is really happening should be the appearance of a small plateau in functions \( U_{i}(f) \) at the peak frequencies. Anyway, this method of ex-core neutron noise analysis may be an useful complement of other methods like VIBREAL and SPEC-4R.

ACKNOWLEDGEMENT

The authors are grateful to Dr. P. Bernard who gave us access to magnetic tapes with recorded noise of the three French PWRs analyzed here, and to Dr. J. Galan for his collaboration during the preparation of the paper.
Fig. 7 The identifying functions $U_i(f)$ for the French F2 PWR neutron noise.

Fig. 8 The identifying functions $V_i(f)$ for the French F2 PWR neutron noise.

Fig. 9 The APSD of the signal of an ex-core detector in arbitrary units (French F2 PWR).

Fig. 10 The APSD of the signal of an ex-core detector in arbitrary units (French F3 PWR).
Fig. 11 The identifying functions $U_i(f)$ for the French F3 PWR neutron noise.

Fig. 12 The identifying functions $V_i(f)$ for the French F3 PWR neutron noise.

REFERENCES


SIGNAL MODELLING

Session chairman: M. Antonopoulos-Domis (Greece)
SUMMARY OF SESSION

All seven presentations in this session deal with time series linear model analysis, namely: univariate AR, MAR and ARMA models. This reflects the increasing interest over the last decade in using such methods for reactor surveillance, malfunction detection and diagnosis. Two of the papers present AR applications on non-stationary data while the rest of the presentations deal with stationary data.

Kishida and Yamada presented separation rules of AR poles, which can distinguish between system and ring poles, using properties of geometrical pole location of AR model. He presented the application of these rules on (a) the artificial JFDR data from the benchmark test of SMORN-III and (b) Borselle reactor noise data.

Ciftcioglu et al. presented a study for optimum choice of the parameters of MAR modelling. The study shows that the finite sampling interval may cause false correlations between the noise sources, assumed to be independent. By application of MAR analysis on data from a digital simulator of point reactor with simplified thermohydraulic feedback the authors find that the covariance matrix becomes increasingly non-diagonal as the sampling frequency decreases. However, they find difficult to estimate the precise role of sampling frequency, due to interdependence of the model parameter. Despite the dependence of parameters on the particular application, they find that memory time and sample-length time criteria are applicable to each individual case commonly for the establishment of optimality.

The objective of the work presented by Hayashi et al. is to test MAR model fitting for system identification, using analog computer-simulated data of a 2-D feedback system. This study of the effect of sampling conditions shows dependence of the results on sampling frequency. It is also found that, although MAR gives correct estimates of power spectra, it does not necessarily give correct estimates of transfer functions. An example of the application of system identification using MAR modelling of the noise data from a pulsed reactor, was also presented.

Hayashi et al. presented tests and application of univariate AR analysis of non-stationary data, the advantage of AR over normal FFT being its ability to use small size sample data. By application of AR analysis to data from (a) analog system simulation and (b) reactor noise during shut down operation, they calculated local and instantaneous spectra. They found that instantaneous AR spectra can be used to analyze non-stationary data, if the length of data span used for each instantaneous spectrum is short in comparison with the time constant of the transients.

Kuroda and Suzuki presented a study of Bayesian AR-modelling on non-stationary data, concerned with the fine structure of reactor noise. Both instantaneous and local spectra were computed from univariate modelling of neutron and pressure noise signals from the Borselle reactor and temperature noise signals from the Phoenix reactor. The local structure of these signals was thus identified.

Nagy et al. presented the signal processing hardware in the Noise Surveillance System and the methodology and experience of application of parametric signal processing and spectral analysis at Paks nuclear power station. The methods include MAR and ARMA modelling. A method for decomposition of PSDs was also applied, to study the effect of common source noises in multivariate situations.
Veres et al. presented a study of the relations among three methods, namely (a) signal transmission path analysis (b) the Dynamic Data System analysis and (c) the linear dependence and feedback analysis, all of which are some linear models of the system, such as MAR or ARMA models. Extending these methods the authors propose the signal effect analysis, defining two new concepts for studying the effect of source noises on variables and intereffects between variables. They illustrate this by analysing reactor pressure and incore neutron fluctuation processes.

In conclusion, the session demonstrated the advantages of AR modelling, namely: capability to analyse cause and effect, applicability to feedback systems and ability to use sample data of smaller size, as compared to classical FFT techniques. The session demonstrated also limitations and unresolved issues namely: dependence of results on the signals selected and the problem of optimum selection of the interdependent model parameters. The dependence of these parameters on the nature of the signals and system dynamics appears to be stronger than that of conventional FFT analysis. It is the reviewer's opinion that a necessary precondition for the acceptance of these models for practical application by the user, is the existence of clear and reliable selection rules for the model order and parametrization. It is also clear that understanding of the physical system is necessary and a black-box approach does not seem to be always feasible. A combination of physical and black-box modelling is perhaps a reasonable option.
POLES AND THEIR WEIGHTS OF AR TYPE MODEL

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ABSTRACT

A separation rule of AR poles is proposed in system identification of reactor noise with AR model. Weights of AR poles corresponding to some of system poles are constant for model order change. On the other hand, weights of AR poles corresponding to ring (or circular) poles decrease in inverse proportion to AR model order. These properties of weights can be utilized for distinguishing between system poles and ring poles equivalent to zeros.

KEYWORDS

Pole location rule; Separation rule; AR model; AR pole; Weight of pole; System pole; Ring pole; AR-MA zero; Contraction of information

1. INTRODUCTION

One of main approaches in reactor noise analysis is AR (autoregressive) model fitting because of its simple recursion algorithms. The AR model is a linear equation, and then the problem of eigenvalues and eigenvectors is crucial from a mathematical viewpoint. The examination of eigenvalues in linear equations corresponds to that of poles in AR models. Rules of asymptotic pole locations of AR model have been proposed in the previous papers (Kishida, Yamada, Bekki, 1985, Yamada, Kishida, and Bekki, 1986, Kishida, Yamada, and Bekki, 1987a, 1987b). Asymptotic pole locations of AR model are classified into four types: (1) multiple ring (or circular) poles equally spacing on a convergence circle, (2) some of system poles inside the convergence circle, (3) non-robust singular pole, (4) robust singular poles. Here the convergence circle is the one centered at the origin with the radius of the modulus of AR-MA zero nearest to the unit circle, and AR-MA zeros are needed for equivalent representation of physical process. We call these classifications a pole location rule of AR model. Since a problem of pole location of AR model is now clear, there remains a problem of AR model which corresponds to eigenvectors in linear equations. Eigenvectors in linear equations correspond to weights of AR poles in AR models. The property of AR weights can be utilized for distinguishing between system poles and ring poles. When we want to apply AR model fitting for time series data obtained from nuclear reactors, it is important to distinguish between system poles and ring poles, since system poles contain dynamical properties and/or physical meaning of nuclear reactors. The property of AR-pole weights is discussed as a pole-separation rule in the present paper.

Before discussing the problem, we summarize the basic equations and ideas mentioned in the previous papers (Kishida 1982a, 1982b, 1984). First we review contraction of information and an equivalent process described by time series models in this chapter. In the next chapter we summarize poles and zeros of time series models and show briefly rules of AR poles location. Then we will discuss a separation rule of AR poles by using properties of geometrical pole location of AR model.
Under the conditions; (a) Markovian process, (b) normal scaling, (c) Gaussian property, and (d) steady operation, macroscopic systems such as nuclear plants are physically described (Kishida 1980) by the linear equation,

$$x(n)=\Phi x(n-1)+f(n),$$

where $x(n)$ is a d-dimensional state vector at discrete time $n$, $f(n)$ a d-dimensional white random force with its d×d variance matrix $V$, and $\Phi$ a d×d constant regression matrix.

Though a sufficiently large number of state variables are usually needed for the Markovian representation of the plant, less than that of them are observed in practical cases. This is the first type of contraction of information, and the contraction of state variables from d to q is described by the observation equation with measurement noise $w(n)$:

$$y(n)=Hx(n)+w(n), \quad (q<d, \text{rank } H=q)$$

where $y(n)$ is a q-dimensional observable state vector, $H$ is a q×d observation matrix, and $w(n)$ is a q-dimensional white random force independent of the system noise $f(n)$ with its q×q variance matrix $W$. By projection of state variables to observable time series space, Eqs. (1) and (2) are rewritten as

$$\begin{cases}
x(n|n)=\Phi x(n-1|n-1)+K\gamma(n) \\
y(n)=Hx(n|n)+IHK\gamma(n),
\end{cases}$$

where the conditional variables $x(n|n)=E(x(n)|Y(n))$ with $Y(n)=(y(n)^T, y(n-1)^T, y(n-2)^T, \ldots)^T$, the innovation $\gamma(n)=y(n)-y(n|n-1)$, the Kalman gain $K=PHI^{-1}$ with $I=PHI^TW$, P is a stable solution of the Riccati type equation $P=\Phi(P-PHI^{-1}HP)^T+V$, and I is the unit matrix. Using the so-called system matrix

$$S(z^{-1})=\begin{bmatrix}
\phi z^{-1}I & K \\
H & I-HK
\end{bmatrix},$$

Eq. (3) is expressed as

$$S(z^{-1})\begin{bmatrix}
x(n|n) \\
y(n)
\end{bmatrix} = \begin{bmatrix}
0 \\
y(n)
\end{bmatrix},$$

where $z^{-1}$ is the time shift operator, $z^{-1}y(n)=y(n-1)$. Multiplying both sides of Eq. (5) by a unimodular matrix of polynomials in $z^{-1}$, which brings no change on both the output $y(n)$ and the input $\gamma(n)$, we obtain an equivalent process under the controllability and observability conditions;

$$\begin{bmatrix}
I & N(z^{-1}) \\
0 & -A(z^{-1}) B'(z^{-1}) \\
0 & I-HK
\end{bmatrix}\begin{bmatrix}
x_1 \\
x_2 \\
\gamma(n)
\end{bmatrix} = \begin{bmatrix}
0 \\
0 \\
y(n)
\end{bmatrix},$$

where $A(z^{-1})$ and $B'(z^{-1})$ are q×q polynomial matrices and $N(z^{-1})$ is a (d-q)×q polynomial matrix. Eliminating unobservable variables $x_1$ and $x_2$ in the above equation, we also have an AR-MA process

$$A(z^{-1})\gamma(n)+B(z^{-1})y(n),$$

where $B(z^{-1})=B'(z^{-1})A(z^{-1})(I-HK)$. Therefore the poles and zeros of AR-MA process are defined from

$$\text{AR-MA (or system) poles: } \det A(z^{-1})=0 \text{ or } \det(\phi z^{-1}I)=0$$

$$\text{AR-MA zeros: } \det B(z^{-1})=0 \text{ or } \det S(z^{-1})=0.$$
2. RULE OF AR POLE LOCATION

In the reactor noise analysis, an AR model is often used (Bernard, and Nagel) because of its fast recursion algorithm. Then, let us fit an AR-MA process (6) by an AR type model. The AR model is of all pole type, though the AR-MA model equivalent to physical processes has not only poles but also zeros. Therefore, AR-MA zeros are transformed into equivalent poles in AR model fitting. This is the second type of contraction of information. In the previous papers (Kishida, Yamada, Bekki, 1985; Yamada, Kishida, and Bekki, 1986; Kishida, Yamada, and Bekki, 1987a, 1987b), we have shown numerically or analytically that the relationship between AR poles and AR-MA poles and/or zeros. Let us review the rule of AR pole location in identification of a univariate AR-MA(1,1) process,

\[ \text{A}(z^{-1})y(n) = \text{B}(z^{-1}) y(n), \]  

with \( A(z^{-1}) = 1 - az^{-1} \) and \( B(z^{-1}) = 1 - bz^{-1} \) \((0 < a < 1, 0 < b < 1)\). Here it should be noted that the AR-MA process (9) is the simplest example in the case of system with observation noise. We treat the AR-MA(1,1) process (9) as a fundamental model in the present paper in order to show the essence of the rule of AR pole location.

Therefore, the system (or AR-MA) pole is given by \( a^{-1} \) and the AR-MA zero is \( b^{-1} \). A univariate AR model of degree \( m \) is defined by

\[ \text{A}(m, z^{-1})y(n) = c(n), \]  

where \( A(m, z^{-1}) \) is a polynomial of degree \( m \) in \( z^{-1} \) and its coefficients are obtained numerically from the Levinson algorithm. Therefore, the \( m \) poles of AR model are defined as the zeros of AR poles: \( A(m, z^{-1}) = 0 \).

Since AR coefficients are determined by nonlinear transformations from physical parameters, it is intractable to examine analytically poles of AR model. Therefore an AR type model defined through the Taylor series expansion of the AR-MA model is considered as a standpoint to explain the pole location of the univariate AR model. The characteristics of AR-pole location are mainly due to the Taylor expansion of MA part of AR-MA model;

\[ A(m, z^{-1}) \approx (1 - az^{-1})^m \left( \sum_{j=0}^{m-1} (bz^{-1})^j \right). \]  

This has been shown rigorously in the papers (Kishida, Yamada, and Bekki, 1987a, 1987b). The outline of the rule of AR model is as follows: Taking an appropriate circle in the complex plane, and then using Rouche's theorem, we can calculate the number of poles of AR type model inside the circle, since

\[ \frac{1 - az^{-1}}{1 - bz^{-1}} = (1 - az^{-1})^m \left( \sum_{j=0}^{m-1} (bz^{-1})^j \right) + (1 - az^{-1})^\infty \left( \sum_{j=m}^{\infty} (bz^{-1})^j \right), \]  

or

\[ (1 - bz^{-1})A(m, z^{-1}) = 1 - az^{-1} - (1 - az^{-1})(bz^{-1})^m \quad \text{with} \quad |bz^{-1}| < 1. \]

For a sufficiently large AR model order, \(|1 - az^{-1}| \gg |(1 - az^{-1})(bz^{-1})^m|\) inside the convergence circle defined by \(|1 - bz^{-1}| = 0\), and conversely \(|1 - az^{-1}| \ll |(1 - az^{-1})(bz^{-1})^m|\) outside the convergence circle.

Therefore, for \( a^{-1}c^{-1} \), the AR model with the order \( m \) has both one pole which corresponds to the original system pole inside the convergence circle and \( m-1 \) poles in the circular domain which contains the convergence circle. This ring poles of the AR model are needed for equivalent representation of the original AR-MA zero. On the other hand, for \( a^{-1}b^{-1} \), the AR model with the order \( m \) has \( m \) poles in the circular domain and no system pole. Hence we have the pole location rule of AR model with a sufficient large model order. The AR model has both the system pole inside the convergence circle and ring (or circular) poles as in the pole location rule. Pole locations of AR model in more practical situations are given in similar manners.

When the modulus of AR-MA pole is almost the same as that of AR-MA zero, it is difficult to determine whether each pole of AR model is the system pole or not without knowledge of the original system (9). In this case, we cannot find a way to distinguish among AR poles on account of calculation errors or statistical errors. In application of the AR model to reactor
control or diagnosis it is important to find a way to distinguish system poles from ring poles, since system poles contain dynamical properties of system. We will show a separation rule of AR poles in the next chapter.

3. SEPARATION RULE OF AR POLES

As mentioned in Introduction, the AR model is described by a linear equation which characteristics are represented by eigenvalues and eigenvectors from a mathematical point of view. Though the examination of eigenvalues or pole location of AR models is clear, there still remains a problem of AR model which corresponds to eigenvectors in linear equations. So, let us treat the problem by examining weights of AR poles in the AR transfer function. From (12) we have the partial fraction expansion

\[
\frac{1}{A(m,z^{-1})} = \frac{c_p}{1-az^{-1}} + \frac{1-bz^{-1}}{1-(bz^{-1})^m} c(z^{-1}) = \frac{c_p}{1-az^{-1}} + \sum_{j=1}^{m-1} \frac{u_j}{1-bw_j z^{-1}},
\]

(13)

where \(c(z^{-1})\) is a polynomial of degree \(m-2\) in \(z^{-1}\) with coefficients \([c_i, i=0, 1, 2, \ldots, m-2]\),

\[w_j = \exp(i2\pi j/m)\] for \(j=1, 2, \ldots, m-1\). Here we have introduced the polynomial \(c(z^{-1})\) in order to evaluate analytically \(c_p\) and \(u_j\). Parameters \(c_p\) and \(u_j\) are weights of the system pole and ring poles, respectively. Each parameter is the coefficient of each mode of the impulse response. Equating the coefficients of equal power in \(z^{-1}\) of both sides of Eq. (13) after reduction of the common denominator, we have the relations

\[
\begin{align*}
  z^0: \quad & 1 = c_p + c_0 \\
  z^k: \quad & 0 = c_p b^k + c_k - c_{k-1} a \\
  z^{m-1}: \quad & 0 = c_p b^{m-1} - a c_{m-2}
\end{align*}
\]

(14)

or in a matrix representation we have

\[
\begin{pmatrix}
  M_{11} & M_{12} \\
  M_{21} & M_{22}
\end{pmatrix}
\begin{pmatrix}
  c_p \\
  C
\end{pmatrix}
= \begin{pmatrix}
  1 \\
  0_{m-1}
\end{pmatrix},
\]

(15)

where \(C := (c_0, c_1, c_2, \ldots, c_{m-2})^T\), \(0_{m-1}\) is the \(m-1\) dimensional zero vector, \(M_{11} := I\), \(M_{12} := (1, 0, \ldots, 0)\), \(M_{21} := (b, b^2, b^3, \ldots, b^{m-1})\), and the \((m-1) \times (m-1)\) matrix \(M_{22}\) is

\[
\begin{pmatrix}
  -a & 1 & 0 \\
  -a & 1 & \ddots & \ddots \\
  0 & -a & 1 & -a
\end{pmatrix}
\]

Solving the simultaneous linear equations (15), and using the inversion formula of block matrix, we have

\[
c_p = (M_{11}^{-1} M_{12} M_{22}^{-1} M_{21})^{-1}
\]

\[
C = M_{22}^{-1} M_{21} c_p,
\]

(16)

That is, putting \(b/a = t\), we have

\[
c_p = (1 - [-a^{-1}, -a^{-2}, \ldots, -a^{-(m-1)}]) \begin{pmatrix}
  b \\
  b^2 \\
  \vdots \\
  b^{m-1}
\end{pmatrix} = \frac{1-t}{1-t^m}
\]

(17)
or \[ c_k = b^m k \sum_{j=1}^{m-k-1} (a^{-1}b)^j c_p, \] for \( 0 \leq k \leq m-2 \)

(18)

since the \((i,j)\) element of \(M_{22}^{-1}\) is \(-a^{-(i-j)-1}\) for \(j \geq i\) and 0 for \(j < i\). Here the weight of the system pole, \(c_p\), is asymptotically 1-t for \(a^{-1}b^{-1}\) and 0 for \(a^{-1}b^{-1}\). This means that the system pole outside the convergence circle has no contribution to the impulse response. Next, let us decompose the second term in Eq. (13) into partial fraction in order to determine the weights of ring poles. Taking advantage of geometrical patterns of ring poles, we can evaluate \(\{u_k\}\) analytically. That is,

\[
\begin{align*}
 u_k &= \left(1-b^{-1}w^{-1}_k\right) \frac{1-bz^{-1}}{1-(bz^{-1})^m} c(z^{-1}) \bigg|_{z^{-1} = b^{-1}w^{-1}_k} \\
 &= \frac{c(b^{-1}w^{-1}_k) \prod_{j=1}^{m-1} w^{-1}_j}{\prod_{j=1}^{m-1} (1-w^{-1}_j)} \cdot \frac{m-1}{j=k}.
\end{align*}
\]

(19)

Since the solutions of equation, \(z^{m-1} = 0\), are \(1, w_k, k = 1, 2, \ldots, m-1\), we have

\[ z^{m-1} = (z - 1)(z - w_1)(z - w_2) \ldots (z - w_{m-1}). \]

Dividing the factor \((z-1)\) on both sides if \(z \neq 1\), we have the relation

\[ z^{m-1} + z^{m-2} + z^{m-3} + \ldots + z + 1 = (z - w_1)(z - w_2)(z - w_3) \ldots (z - w_{m-1}). \]

(20)

Putting \(z = 1\) on both sides of Eq. (20), we obtain the formula

\[ (1-w_1)(1-w_2)(1-w_3) \ldots (1-w_{m-1}) = m. \]

(21)

The numerator of relation (19) is \(-c(b^{-1}w^{-1}_k)w^{-k}_1\). Using the formula (21) we calculate the denominator as \(w^{-k}_1 (m-2)_m / (1-w^{-1}_1)\). Therefore we have the weights of ring poles

\[ u_k = c(b^{-1}w^{-1}_k)(1-w^{-1}_1)/m. \]

(22)

Let us evaluate \(c(b^{-1}w^{-1}_k)\) in the typical case, \(k = 1\), since the other cases can be treated in the same manner. From Eq. (18) we have

\[
\begin{align*}
c(b^{-1}w^{-1}_1) &= \sum_{j=0}^{m-1} b^j \sum_{k=1}^{m-j-1} (a^{-1}b)^k c_p (b^{-1}w^{-1}_1)^j \\
&= \sum_{j=0}^{m-1} \frac{1 - (\frac{b}{b^{-1}w^{-1}_1})^j}{1 - \frac{b}{b^{-1}w^{-1}_1}} c_p \frac{b}{b^{-1}w^{-1}_1}. \]
\end{align*}
\]

(23)

Then we can evaluate a weight of a typical ring pole as

\[ u_1 = \frac{b}{b^{-1}w^{-1}_1} \frac{1}{m} = \frac{1}{m}. \]

(24)

Similarly one can show that the weights of ring poles decrease in inverse proportion to the AR model order.

Comparing the AR weight of system pole, Eq. (17), with the AR weight of typical ring pole, Eq. (24), we can find that the weight corresponding to system pole is invariant to the AR model order but the weight corresponding to typical ring pole decreases in inverse proportion to the AR model order.
order. These properties provide us a method for separation of AR poles. These properties hold in a scalar or vector AR-MA process with higher model orders, and the generalization of this rule to vector AR-MA model with higher orders will be reported in forthcoming papers. Therefore we call these properties the separation rule of AR poles. That is, if the weight of an AR pole is constant for model order change, this pole is considered to be a system pole. And if the weight of an AR pole decreases in inverse proportion to the AR model order, this pole is considered to be one of ring poles.

4. AR WEIGHTS IN REACTOR NOISE

Based on the rules discussed above, we examine weights of AR models fitted for real time series data. One example is the artificial JPDR data distributed for the benchmark test of SMORN-III, and the other is the Borssele data distributed in SMORN-IV. The JPDR analog data on the channel 2 (noise data of vessel pressure) were A/D converted at the sampling frequency of 100 Hz through a second order delay filter. The autocorrelation function was computed by using the inverse Fast Fourier Transform (FFT) method from 65 blocks (4096 of time series data per block). On the other hand, the Borssele reactor noise data distributed were recorded in digital form of 64 Hz sampling. 1140 blocks of 512 data were used for the autocorrelation function by the same method. After obtaining each AR model by the Levinson-Durbin algorithm, we have calculated AR poles with the Bairstow method at Osaka University on a NEC ACOS-1000 computer with use of double precision of 72 bits, and finally obtained weights of AR poles (Sugibayashi, 1986).

4.1 Artificial JPDR-II data

Poles of AR model of order 14 are shown in Fig. 1. The AR pole, a, in Fig. 1 is considered to be a system pole, since it is inside a convergence circle. On other hand the poles from c to h are ring poles, since they are equally spaced on the convergence circle. From the pole location we cannot judge whether the pole b is a system pole or one of ring poles. Then we calculate weights of AR poles from a to h, and they are shown in Fig. 2 for AR model order from 15 to 30. The weights of AR poles a and b are converging to some constants for increasing AR model order. The
weights of two AR poles are dominant in comparison with the others as in Fig. 2. Then we found that the AR pole b is also a system pole. The other weights of the ring poles from c to h are inversely proportional to AR model order, and the magnitudes of them are of the same order. Hence we confirmed that two AR poles a and b are system poles and that the other poles from c to h are ring poles. In other words the behaviors of pole-weights are the same as those of system poles and ring poles discussed in chapter 3.

4.2 Borssele reactor noise data

Fluctuations of in-core neutron detector (IN-14) in the Borssele reactor were analyzed by the AR modeling. Since the signal of the detector includes the information of core barrel motions, tube motions and so forth, the dynamics of AR model is expressed by not only AR poles of neutron dynamics but also AR poles of these mechanical motions. The pole location of AR model with order 14 is shown in Fig. 3. The structure of pole location of this system is more complicated than that in Fig. 1. Weights for AR poles from a to h are shown in Fig. 4, which indicates that the pole a is a system pole, since the weight of the pole a is large and has no tendency of inverse proportionality to AR model order. On the other hand, weights of the poles from d to g have decreases inversely proportional to the AR model order, and then they are considered to be ring poles. However, there remains a problem in determination of poles b and c. The weights of both poles have slightly small decreases in comparison with those of ring poles from d to g. Then both poles b and c can be considered as hybrid poles between system and ring poles. It is considered that this phenomenon of weight occurs in the case that there are system poles outside the convergence circle. Further examinations of hybrid poles are needed.

In Fig. 3 there is a real AR pole near the system pole a. The AR pole bifurcates into a pair of complex AR poles, when the AR model order varies. This is due to the complexity of real system. We also find a nonrobust singular pole (Kishida, Yamada, and Bekki, 1987a) in the higher AR model order. In Fig. 4 there is a peak in the weight of the system pole a in the lower AR model orders. The mechanism of this peak is also a open problem.

Fig. 3 Pole location of AR model with order 14 fitting Borssele neutron data (IN-14).

Fig. 4. Weights of AR poles in Borssele neutron data.
Since our separation rule of AR poles is based on the asymptotic properties of AR model, we must pay our attention to accuracy of numerical calculations and convergence of AR model in practical situations. In the case that there is a system pole near the unit circle, the AR pole corresponding to the system pole does not approaches quickly to the system pole. In this case, we have a possibility of non-constant behavior that a weight of AR pole corresponding to a system pole varies with the AR model order, though the asymptotic behavior of the weight is theoretically constant. We also have an extrapolation error of correlation function in a lower frequency region as in aliasing phenomenon in a higher frequency region. These related problems will be discussed in forthcoming papers.

5. CONCLUSION

We summarize the asymptotic properties of AR pole location (AR pole location rule);
1) system (AR-MA) poles inside the convergence circle,
2) multiple circular or ring poles which are equivalent to some of AR-MA zeros,
3) robust and non-robust singular poles outside the convergence circle.
The other system poles and AR-MA zeros which are not listed up in the above pole location of AR model are equivalently expressed by distortions of the three types of pole locations.

As to system poles and ring poles, in addition, we can summarize the separation rule of AR poles:
1) If the weight of an AR pole is constant for AR model order change, the AR pole is a system pole.
2) If the weight of an AR pole is inversely proportional to the AR order, the AR pole is one of ring (or circular) poles.

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REFERENCES

STUDIES ON MULTIVARIATE AUTOREGRESSIVE ANALYSIS USING SYNTHESIZED REACTOR NOISE-LIKE DATA FOR OPTIMAL MODELLING

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Abstract - Studies on the multivariate autoregressive (MAR) analysis are carried out for the choice of the parameters for modelling the data obtained from various sensors optimally. Accordingly, the roles of the parameters on the analysis results are identified and the related ambiguities are reduced. Experimental investigations are carried out by means of synthesized reactor noise-like data obtained from a digital simulator providing simulated stochastic signals of an operating nuclear reactor so that the simulator constitutes a favourable tool for the present studies aimed. As the system is well defined with its known structure, precise comparison of the MAR analysis results with the true values is performed. With the help of the information gained through the studies carried out, conditions to be taken care of for optimal signal processing in MAR modelling are determined. Although the parameters involved are related among themselves and they have to be given different values suitable for the particular application in hand, some criteria, namely memory-time and sample length-time play an essential role in AR modelling and they are found to be applicable to each individual case commonly, for the establishment of the optimality.

1. INTRODUCTION

In the last decade, considerable attention has been paid to the application of time-series analysis methods to nuclear reactors and as result of these studies, a number of implementations concerning reactor surveillance, malfunction detection and diagnosis have been reported (Fukunishi, 1977; 1978; Matsumura et al., 1978; Kitamura et al.; Bertusi et al., 1980; Upadhyaya and Kitamura, 1981; Suzuki et al., 1986). With reference to various inherent noise sources in an operating reactor, stochastic modelling of the noise signals originating from these noise sources occurs to be rather appropriate for the above mentioned goals. In particular, in nuclear reactors the method of autoregressive (AR) modelling possesses some desirable features over some other stochastic modelling techniques such as moving average (MA) and autoregressive moving average (ARMA) modelling. Among these features mention may be made to the computing efficiency and simplicity.

By applying a multivariate AR process to measured noise signals from the system in hand for analysis, the dynamics of the system can be identified. In the modelling, the system is assumed to be excited by the inputs which might be immeasurable and mutually uncorrelated as is often the case in an operating reactor. Though the modelling procedure is apparently straightforward, in practice care should be exercised for an optimal choice of the modelling parameters, i.e., sampling period, model order and sample length which is equal to a predetermined number of data points as a fraction of the total sampled data called sample size in this context. In particular, the optimality is, in this context considered as follows. According to the diagonal covariance matrix criterion, minimization of the false correlations which appear between the noise sources assumed to be independent in the process, is aimed and some relevant quantitative measures and model parameters on which a design is based, are established. In this way, the non-diagonalization of the covariance matrix can be considered as a penalty function subject to minimization for the optimality being sought.
2. LINEAR SYSTEM ESTIMATION BY MAR MODELLING

Although the multivariate autoregressive (MAR) modelling constitutes a powerful method, application to nuclear reactors is confined almost to the last decade. Generally speaking, in analysis of a noisy physical system these variations can be modelled for a group of \( m \) signals in the form

\[
\mathbf{y}_t = \sum_{i=1}^{p} \mathbf{A}_i \mathbf{y}_{t-i} + \mathbf{n}_t
\]

(1)

where \( \mathbf{y}_t \) is \((m \times 1)\) signal vector related to process signals at time \( t \); \( \mathbf{A}_i \) \((i=1, 2, \ldots, p)\) \((m \times m)\) matrices known as AR coefficients matrices; \( p \) model order ; \( \mathbf{n}_t \) noise source vector, the sources being uncorrelated and white.

The AR matrices can be estimated by solving the generalized Yule-Walker equations of the form

\[
\sum_{i=0}^{p} \mathbf{A}_i \mathbf{R}_{k-i} = 0 \\
\text{for } k=1, 2, \ldots, p, \quad \mathbf{A}_0 = \mathbf{I}
\]

(2)

where \( \mathbf{R}_k \) represents the theoretical covariance functions with delayed number \( k \) of the stationary process \( \mathbf{y}_t \) given by

\[
\mathbf{R}_k = \mathbb{E} (\mathbf{y}_t^T \mathbf{y}_{t-k})
\]

(3)

\( \mathbb{E} \) being the symbol for averaging. In practice, \( \mathbf{R}_k \) is replaced by the estimated covariance function \( \hat{\mathbf{C}}_k \) and the matrix equation (2) can then be solved for the AR matrices \( \hat{\mathbf{A}}_i \). From Eqs. 1 and 3, the noise covariance matrix \( \mathbf{\Sigma} \) is estimated by

\[
\mathbf{\Sigma} = \mathbf{\Sigma}_0 - \sum_{i=1}^{p} \mathbf{A}_i \hat{\mathbf{C}}_i \mathbf{A}_i^T
\]

(4)

where the symbol \( T \) stands for the transpose.

The AR-model can be expressed in the frequency domain in the form

\[
\mathbf{Y}(f) = \sum_{i=1}^{p} \mathbf{A}_i \mathbf{Y}(f) \exp(-j2\pi f at) + \mathbf{N}(f)
\]

(5)

which can be expressed in the form

\[
\mathbf{Y}(f) = \mathbf{G}(f) \mathbf{N}(f)
\]

(6)

\( \mathbf{G}(f) \) being the system transfer matrix given by

\[
\mathbf{G}(f)^{-1} = \mathbf{I} - \sum_{i=1}^{p} \mathbf{A}_i \exp(-j2\pi f at)
\]

(7)

where \( a \) is the sampling period and \( I \) unity matrix.

The spectral matrix \( \mathbf{S}_{yy}(f) \) of the signals is

\[
\mathbf{S}_{yy}(f) = \mathbf{G}(f)^* \mathbf{S}_{\Sigma} \mathbf{G}(f)
\]

(8)

where the symbol * indicates the complex conjugate. Since the noise sources are not correlated among themselves, taking for granted that the covariance matrix \( \mathbf{\Sigma} \) is diagonal, the autospectrum \( \mathbf{S}_{ii} \) of signal \( i \) and the noise contribution ratio (NCR) of noise source \( N_j \) to signal \( Y_i \) are defined by

\[
\mathbf{S}_{ii}(f) = \sum_{j=1}^{m} \mathbf{G}_{ij}(f)^* \mathbf{G}_{ij}(f) \mathbf{S}_{jj}
\]

(9)

and

\[
\text{NCR}_{ij}(f) = \frac{\mathbf{G}_{ij}(f)^* \mathbf{G}_{ij}(f) \mathbf{S}_{ij}}{\mathbf{S}_{ii}(f)} = \frac{|\mathbf{G}_{ij}(f)|^2 \mathbf{S}_{ij}}{\mathbf{S}_{ii}(f)}
\]

(10)
respectively. If \( \Sigma \) is not diagonal, \( S_i(f) \) differs from the expression above and also \( \Sigma \) NCR, \( \Sigma \) differs in general from unity as well.

3. INVESTIGATIONS RELATED TO THE INTERDEPENDENCE OF THE PARAMETERS IN AR MODELLING

3.1. Effect of sample size on the model order selection through AIC

In AR modelling, briefly described above, the model order \( p \) is generally speaking not known and depends on the system dynamics. From information theoretical considerations (Akaike, 1974) the value of \( p \) is obtained when the function

\[
AIC(p) = Nfn \left| \frac{2}{N} \right| + 2\sigma^2 p
\]

becomes minimum for \( p \). Above, \( N \) is the sample size, i.e. total number of samples used in the covariance estimation; \( \left| \frac{2}{N} \right| \), the determinant of the noise covariance matrix; \( \sigma \), the number of process variables. Although in the equations representing the AR model the sample size parameter \( N \) does not appear, its role becomes obvious, in particular when the AIC is adopted for optimum model order determination.

To study the dependence more quantitatively let us consider the univariate case in which case we have from Eq.11 for the AIC to be minimum

\[
\frac{\sigma^2}{\sigma^2} = \frac{\sigma^2}{N}
\]

(12)

where \( \sigma^2 \), \( \sigma^2 \) are the sample variances of gaussian zero-mean white noise sequences. Therefore we define chi-square variables \( \chi^2 \), \( \chi^2 \) of the form

\[
\chi^2_1 = \frac{\sigma^2}{\sigma^2}
\]

(13)

\[
\chi^2_2 = \frac{\sigma^2}{\sigma^2}
\]

(14)

with respective degrees of freedoms \( v_1 = N-p \), \( v_2 = N-p-1 \) and \( \sigma^2 \) is the population variance. From Eqs.(13) and (14) we form

\[
\chi^2 = \chi^2_1 - \chi^2_2 = \frac{(\sigma^2 - \sigma^2)}{\sigma^2}
\]

(15)

where \( \chi^2 \) is a new \( \chi^2 \) variable with \( v = v_1 - v_2 = 1 \) degrees of freedom and independent of the variables \( \chi^2_1 \) and \( \chi^2_2 \). Therefore

\[
F = \frac{\frac{\sigma^2}{\sigma^2}}{\sigma^2}\frac{\sigma^2}{\sigma^2}/(N-p-1)
\]

(16)

is a F distribution variable with \( v_1 = 1 \) and \( v_2 = N-p-1 \). Hence from Eq.12, the model order is adopted for

\[
F = \frac{\sigma^2}{\sigma^2}\frac{\sigma^2}{(N-p-1)} < \frac{2(N-p-1)}{N} = 2
\]

(17)

which corresponds to the F-test with \( v_1 = 1 \), \( v_2 = N-p-1 \) degrees of freedom. As a particular example \( v_1 = 100 \) and 2.5% critical value \( F_{0.025} \) is found to be approximately \( F_{0.025} = 5 \) which indicates that \( F < 2 \) determined according to AIC is not significant provided 2.5% critical value in F-test adopted. Above \( F = 2 \) approximately corresponds to \( F = 0.17 \). For the multivariate case with \( m \) signals, inequality (17) takes the form

\[
F = \frac{\left| \frac{2}{N} \right|}{\left| \frac{2}{N} \right| + 1} \frac{(N-m^2)}{m^2} < \frac{2(N-m^2)}{N}
\]

(18)

From the viewpoint of F-test and 2.5% critical value, AIC is, apparently, rather strict in determining the model order. Using this criterion it becomes probable that an excessive value for model order \( p \) is adopted yielding some degradation in both computing efficiency and system estimation. Hence a better criterion might be the modified AIC

\[
AIC M(p) = N\ln \left| \frac{2}{N} \right| + F_{a^2} \sigma^2 p
\]
with α the adopted uncertainty in the F-test.

3.2. Effect of sampling frequency on the noise covariance matrix

The diagonal nature of the noise covariance matrix \( \Sigma \) is an important property to be made use of for verifying the accuracy of the system estimation results through MAR modelling. Since in the modelling procedure noise sources driving the system are uncorrelated, any virtual correlation between any pair of sources is to be considered false. On the other hand, in practice, \( \Sigma \) is reported to be non diagonal (Upadhyaya et al, 1980) yielding inaccurate results. The cause of the non-diagonal character can be some strong mutual interaction among the noise sources that it implies the presence of a hidden variable which constitutes a common input to more than one observed variable (Kishida and Sasakawa, 1980). To obtain insight into the cause of the non-diagonalization the ensuing considerations relevant to AR modelling become imperative.

In real systems, due to its continuous time character, although apparent information transmission between the signals takes place, there is always a certain lapse of time necessary for information transmission between input and output signals. The entire signal transmission can better be expressed in terms of the time constants of the system. On the other hand, because of lack in the continuity, the AR modelling becomes an approximate approach for the analysis of such systems. That is, the physical process is not an AR process but is forced into such a model. The delay effect in AR modelling can be seen from Eq.1 where the effects of variations in one signal \( y_i \) at time \( t \) cannot be transferred to the other signals before time \( t + \Delta t \). The model does not allow for immediate response of \( y_i \) to \( y_j \). Strictly speaking it does not allow any response at all before time \( \Delta t \) elapsed. This effect is independent of the noise signal frequency content, although the effect is less for signals having a smaller frequency bandwidth. For physical systems, there can easily be some response within this time. Fig.1 illustrates a signal \( y_1 \) responding to a first order system on a given input \( y_i \). If the sampling instants are not synchronous with the input variations, a variation in the sampled version of \( y_i \) and \( y_j \) will occur simultaneously. It is clear that such variations of signals cannot be modelled as

\[
\chi(t) = \sum_{i} A_i y(t-i\Delta t)
\]

if index parameter \( i \) is always greater than zero. This is because the modelling does not take into account the instantaneous response although the variation in Fig.1 will automatically be interpreted as an immediate fluctuation in the signal \( y_2 \). Since the causality
principle in the physical systems is violated by the immediate response, correlations in the
modelled matrix take place. The case can be expressed in mathematical form as follows.
Inclusion of the i=0 term in Eq. 4 would improve the model to allow for the instantaneous
response, i.e.,
\[ Y_t = \sum_{i=0}^{p} \delta_i Y_{t-i} + n_t \tag{19} \]

The noise source vector \( n_t \) is a white noise sequence relevant to the diagonal covariance \( \Sigma \)
matrix. Eq. 19 can be cast into a standard MAR modelling form by removing \( \delta_0 \) term from the
summation, namely
\[(I-\delta_0)Y_t = \sum_{i=1}^{p} \delta_i Y_{t-i} + n_t \]
or
\[ Y_t = \sum_{i=1}^{p} (I-\delta_0)^{-1} \delta_i Y_{t-i} + (I-\delta_0)^{-1} n_t \]
and hence
\[ Y_t = \sum_{i=1}^{p} \beta_i Y_{t-i} + \zeta_t \tag{20} \]

Eq. 20 is a standard MAR model with the noise source vector \( \zeta_t \) that can be identified by the
standard methods. The covariance matrix \( \Sigma' \) is related to the covariance matrix \( \Sigma \) by the
relationship
\[ \text{E}[\zeta_t^T \zeta_t] = \Sigma' = (I-\delta_0)^{-1} \Sigma (I-\delta_0)^{-T} \tag{21} \]

which shows that \( \Sigma' \) is not diagonal. Since these false correlations between the noise
sources might cause serious erroneous analysis results Upadhyaya et al (1980) suggested the
diagonalisation of the noise covariance matrix by means of its eigenvectors which are or-
thonormalized among themselves. We note that the relative noise contribution of the noise
source \( j \) on signal \( i \) is given by Eq. 10 where correlations among the noise sources are
ignored. To evaluate the effect of this approximation the \( \Sigma' \) matrix defined above is trans-
formed as
\[ \varrho = \frac{H}{H}^T \Sigma' \frac{H}{H}^T = \text{diag} [A_1, A_2, A_3, \ldots, A_m] \tag{22} \]

So that the resultant matrix \( \varrho \) becomes exactly diagonal yielding
\[ \Sigma'' = \frac{H}{H}^{-1} \varrho \frac{H}{H}^{-T} \tag{23} \]

which is equivalent to transforming the noise source vector \( n \) as
\[ n' = \frac{H}{H} n \tag{24} \]

Hence the spectral matrix \( S_{yy}(f) \) can be expressed as
\[ S_{yy}(f) = \varrho(f)^{*} \Sigma''(f) = [\varrho(f)\varrho^{-1}]^{*} \varrho^{*} [\varrho(f)\varrho^{-1}]^{T} \tag{25} \]

We define
\[ \varrho(f)\varrho^{-1} = \varrho'(f) \tag{26} \]
so that
\[ S_{yy}(f) = \varrho'(f)^{*} \varrho \varrho'(f)^{T} \tag{27} \]

Accordingly, the auto-spectrum \( S_{ii} \) takes the form
\[ S_{ii}(f) = \prod_{j}^{M} |G_{ij}(f)|^2 \lambda_j \tag{28} \]
and the modified noise contribution ratio denoted by NCR' becomes

$$NCR'_i(f) = \frac{|Q_i|^2 \lambda_i}{S_{ii}}$$  \hspace{1cm} (29)

One might argue that, since the $Q$ matrix is exactly diagonal and if $NCR'(f)$ function is in good agreement with the NCR($f$), we can reasonably infer that the effect of neglecting the off-diagonal elements of $\frac{u'}{u}$ matrix is not significant. However, the difficulty here is that the transformation matrix $H$ is not unique. Because $H$ is constructed in such a way that the elements of the successive columns of $H$ are the elements of the eigenvectors of $\frac{u'}{u}$ and therefore $H$ can be formed in different ways. This explains why $NCR'_i$, in Eq.29 cannot be compared with NCR$_i$, in Eq.10. In particular, in case $H$ is an orthonormal modal matrix of $\frac{u'}{u}$, then the columns of $H$ comprise the elements of real mutually orthogonal unit vectors, i.e., orthonormal unit eigenvectors of $\frac{u'}{u}$ so that the transformation $n' = Hn$ does not change the length of the vector $n$ but change the source strength of each individual noise source, in a general case. Since these changes which provide the diagonalization of $\frac{u'}{u}$ is arbitrary, the related effects on the elements of the system transfer matrix become unpredictable. Consequently $NCR'_i$, in Eq.29 becomes irrelevant to the true NCR$_i$, in Eq.10. As result of this, $NCR'_i(f)$ cannot be used as an alternative to NCR$_i(f)$ in order to verify the validation of NCR analysis. Moreover noise source vector $n'$ has no physical meaning and therefore one might expect deviations between NCR($f$) and the artificial quantity $NCR'_i(f)$.

Validity of NCR can be verified by means of a cumulative NCR analysis as the latter quantity has to amount to unity. Any deviation from unity would imply false correlations between pairs of noise sources.

To reduce the false correlations among the modelled noise sources, the sampling frequency should be increased. By doing so, variation in one signal, in one sampling period due to other signals can be decreased provided that the transfer function between the pair of signals has a limited bandwidth as this is the case for physical systems.

To see the implication of increasing sampling frequency in relation to the residual noise variance, we consider a first-order AR process for simplicity. The continuous parameter counterpart is given by

$$\frac{a \Delta t}{1 - a} \dot{y}(t) + y(t) = \epsilon(t)$$  \hspace{1cm} (30)

where

$$\tau = \frac{a \Delta t}{1 - a}$$  \hspace{1cm} (31)

is assumed to be finite. Hence, if $\Delta t \rightarrow 0$ then $a \rightarrow 1$. If we consider a stationary, continuous first-order random process given by Eq.30, the corresponding AR process is expressed by

$$y_n - ay_{n-1} = \frac{\Delta t}{\tau} \epsilon(t_n)$$  \hspace{1cm} (32)

where

$$\epsilon_n = \frac{a \Delta t}{\tau} \epsilon(t_n)$$  \hspace{1cm} (33)

is the residual noise. Thus, the residual variance $\sigma_{\epsilon^2}$ in terms of the variance of the noise source driving the first-order AR process, can be expressed by

$$\sigma_{\epsilon^2}^2 = (\frac{\Delta t}{\tau})^2 \sigma_{\epsilon^2}^2 = (\tau \Delta t)^{-2} \sigma_{\epsilon^2}^2$$  \hspace{1cm} (34)

Assuming the inequality $|\Delta t/\tau| < 1$ is satisfied, then we write from Eq.34

$$\sigma_{\epsilon^2}^2 = \tau^2 (\Delta t)^2 \sigma_{\epsilon^2}^2 = \tau^2 \sigma_{\epsilon^2}^2$$  \hspace{1cm} (35)

where $f_s$ is the sampling frequency. Here, it is important to note that, since $\sigma_{\epsilon^2}$ remains constant in a stationary process the variance $\sigma_{\epsilon^2}$ is dependent on the sampling frequency, the dependence being related to the order of the system.

4. EXPERIMENTAL RESEARCH

In order to verify the considerations presented in the preceding section and establish criteria for optimal AR modelling simulation work was carried out. The simulation involved concerns a point nuclear reactor model with simplified thermohydraulic feedback (Van Dam,
Design of the simulator is described elsewhere (Van Dam et al., 1981). The properties of digital simulation from the viewpoint of AR analysis were described before (Ciftcioglu, 1986; Hoogenboom et al., 1986). During the studies, three output signals are considered. These are reactor power (P), fuel temperature (T₁), and the coolant temperature (T₂). Input signals are three noise sources chosen as reactivity noise due to control rod vibrations, noise in the inlet temperature of the coolant acting on the coolant temperature and noise in the coolant flow velocity acting on the fuel temperature. The noise sources were realized using independent pseudo random number generators, providing uncorrelated white noise and appropriate digital shaping filters (Ciftcioglu, 1985a; 1985b) to obtain the desired frequency spectra.

In the simulation a delay which is at least equal to one time step is provided between each pair of process signals for causality. For the case, the delay is equal to both the time step used in the digital computation and the sampling period, the noise covariance matrix is found to be acceptable as diagonal while the normalized off-diagonal elements have magnitude of the order of 10⁻². Also the cumulative NCRs are found to be summing up to unity. This corresponds to a matrix from Eq. 21 being approximately equal to zero as one should expect. On the other hand, with the same conditions except that the sampling period is greater than the delay, the noise covariance matrix is found to be non-diagonal, the non-diagonalization being stronger as the sampling period increases. For this case, the cumulative NCRs (CNCRs) do not amount to unity. The deviations become larger with increasing values of the sampling period. These observations are illustrated in Fig. 2, where the sampling period varies from 5 ms up to 40 ms while the basic time step in the simulation is 5 ms.

![Graph showing NCRs](image)

**Fig. 2. Cumulative NCRs (Symbols ○, □ and -) and sum of NCRs for different sampling period (Δt) (x: 5 ms; +: 10 ms; ○: 20 ms and +: 40 ms).**

Delay (Δt_d) between all pairs of signals is 5 ms.

In addition to the above stated results, the residual noise variances decreased with the increase of the sampling frequency as is seen from Eq. 35. However the precise role of the sampling frequency on the diagonalization is difficult to estimate by the experiments as reasoned below.

The degree of the diagonalization is assessed by means of the normalized noise covariance matrix Σ where the diagonal elements are normalized to unity. Since false correlations among the noise sources are reduced rapidly by the increase of the sampling frequency, the diagonalization improves rather fast in this case. On the other hand, variation in the sampling rate yields different model orders in the MAR analysis, the model order being determined by the AIC. The sample size N being the same, model orders are found to be higher than those obtained with smaller sampling rates. However it is observed that the memory-time necessary to characterize the dynamics of the process takes values smaller than that obtained with smaller sampling rate. As an experimental example, delay between any pair of signals being equal to 5 ms and for sampling period of Δt = 5 ms, the model order is obtained to be p = 90; for Δt = 10 ms, it is 55, N being equal to 1024 x 64 in both cases. Memory-time parameter for each case is 450 ms and 550 ms, respectively. Since the duration characterizing the system dynamics, i.e., memory-time, is found to be different in each case, one has to be cautious about the AIC value in relation to model order, if we assume that there might be some degree of deviation from the optimality of the model order.
in either case. Consequently, this would cause changes in the residual noise source variance so that the exact cause of the change in variance remains unknown. To force the model order in either case to make the memory-time parameter equal would not be the solution to this problem since the deviation from the AIC is by no means known.

In determining the model order according the AIC, the value of AIC starts to converge to a minimum relatively rapid in the beginning. However, especially for large sample size N, with the same value for the model order having been attained, the convergence becomes very slow and the improvement gained in the modelling by the increase of the model order beyond this certain value is apparently not justified. This can be explained by means of the inequality 18. Namely, for \( v_i = 1, v_j = \frac{2}{1000} \) and 2.5% critical value \( \chi^2_{0.025} = 5.02 \) which implies that the case is not significant provided conventional 2.5% critical value for the test of hypothesis is adopted. In other words, the AIC seems, in general, to overestimate the significance of the variation of the variances ratio, although the variation might be rated as non-significant with a reasonable confidence. To get insight into the problem we note that in the derivation of AIC, multiple decision procedure with variance estimation rather than hypothesis testing was considered. However these two procedures are closely related and therefore the consideration of both procedures for a compromise would improve the model order estimation. In essence the trouble lies in that in the AIC, apparently the "penalty term" which takes account of the number of parameters in the AR model is underestimated.

Taking into account the above presented observations, let us write the F-test, given by inequality 18, with 2.5% critical value of the form

\[
\frac{|x|^2}{F_{1, p-1}} - 1 \left( \frac{N-s^2 p-1}{N-s^2} < 5.02 \right) \frac{(N-s^2 p-1)/N}{(N-s^2 p-1)/N}
\]

(36)

where the modification introduced alleviates the strong restriction of the AIC in determining the optimum model order. From above, the AIC is modified to be

\[
\text{AICM} = N \ln |\hat{\Sigma}| + 5.02 N^{-2} p
\]

(37)

the excessive model order with very slow convergence being eliminated. Application of the modified form of the criterion yielded reduction in the model order without at least deteriorating the analysis results obtained with the original form if the results are not improved. However the exact implication of the modified "penalty term" and the precise statement of the improvements obtained, in particular concerning the memory-time parameter, remain subject to further studies. It might be of value to point out that the form of the criterion in Eq. 37, to some extent, corresponds to the BIC criterion (Akaike, 1978; 1979) based on Bayesian modification of the AIC.

From the experiments, it is to conclude that increase in the sampling rate provides improvement in the diagonalization of the \( \hat{\Sigma} \) matrix provided noise sources are uncorrelated. This implies, with relatively large sampling period, the MAR modelling tends to determine the model order too large in order to compensate for the false correlations arising due to instantaneous responses in the system under investigation. This conclusion corroborates the observation that the memory-time parameter is generally smaller than that obtained with larger sampling period, the degree of the difference being dependent on the sampling time interval. The observation above is not to say that a slowly varying signal requires larger model order. Generally it is true that higher sampling rate requires larger model order to obtain a sufficiently close fit of the spectrum with the result that memory-time parameter may, to some extend, be varying for different sampling rates, as reported above.

As result of the experimental research, the following criterion for optimal signal processing in MAR analysis can be used as a rule of thumb, where optimal is referred in relation with both the diagonalization of the noise covariance matrix and the consistent memory-time parameter as explained below.

Sample-length-time parameter \( (n \tau) \), which is defined as the product of the sampling period and certain number of data as fraction of the total sum should be at the order of 10\( \tau \) or greater where \( \tau \) is the largest time constant involved in the process. In mathematical form, this is expressed by

\[
n \tau \geq 10 \tau
\]

(38)

where \( n \) is the sample-length which is equal to a predetermined number of data points forming a data group as a subgroup of the total sum so that \( n \times M = N \) i.e., the total number of observations called sample-size, \( M \) being the number of the data groups formed. In processing the data, each group is processed separately and the results of the processing are averaged over the number of data groups \( M \). For sample-size \( N \), the application of AIC in Eq. 15 (or
rather alternatively AICM in Eq. 37) yields the optimal model order \( p \) so that the memory-time parameter defined as \( \rho_t \) remains to be approximately constant. For the sampling period \( \Delta t \), the sample length having been determined from inequality 38, particular MAR analysis results in connection with the memory-time investigations are reported in Table 1 where delay (\( \Delta t_d \)) between each pair of signals is 5 ms.

<table>
<thead>
<tr>
<th>( \Delta t ) (ms)</th>
<th>( n )</th>
<th>( N = n \times M )</th>
<th>( p )</th>
<th>memory-time (ms)</th>
<th>( p \times \Delta t )</th>
</tr>
</thead>
<tbody>
<tr>
<td>5</td>
<td>1024</td>
<td>1024x64</td>
<td>90</td>
<td></td>
<td>450</td>
</tr>
<tr>
<td>10</td>
<td>1024</td>
<td>1024x64</td>
<td>55</td>
<td></td>
<td>550</td>
</tr>
<tr>
<td>20</td>
<td>1024</td>
<td>1024x64</td>
<td>29</td>
<td></td>
<td>580</td>
</tr>
</tbody>
</table>

As it is seen from Table 1, memory-time parameter is smaller than that obtained with larger sampling period. From the view point of correct sampling period, \( \Delta t = 5 \text{ ms} \) is the most favourable. Rather large difference between the memory-time parameters obtained for \( \Delta t = 5 \text{ ms} \) and \( \Delta t = 10 \text{ ms} \) is partly because of the relatively strong violation of the sampling period provision as explained earlier. The difference becomes smaller for \( \Delta t = 10 \text{ ms} \) and \( \Delta t = 20 \text{ ms} \) as the violation becomes relatively less while it is dependent on the rise-time involved in the system dynamics. On the other hand memory-time parameter increases as the model order tends to be somewhat higher than the optimum value while the sampling period provisions required are not observed. Nevertheless, especially for 10 ms and 20 ms, the results are reasonably consistent.

Incidentally, it might be of value to point out that the dependence of the model order \( p \) on the sample-size \( N \) is evaluated by means of inequality 18 (or alternatively 36) where one sees that the increase of \( N \) requires slightly lower model order as this is dependent on the particular process being modelled. Therefore memory-time may, to some extend, be varying for different sample size. However it is important to note that memory-time is a characteristic value of the process being independent of the sample-length or sample-size as long as inequality 38 holds. Any inconsistency arising beyond some acceptable limits would involve the question of the optimality of the model order.

Investigations indicate that, about the sampling frequency, there is no Nyquist-like criterion for an optimal, strictly speaking minimum, value. This is because the sampling frequency should be as high as possible to take into account the instantaneous response for accurate measurements. An experimental procedure proposed by Kleiss (1983) is as follows.

The successive higher sampling frequencies are tested until a satisfactory degree of diagnolisation is obtained for a specific identification task. This is achieved by the \( P \) matrix and the NCRs obtained. One may argue that sampling rate cannot be increased without penalty, i.e., larger model order as pointed out above. However this is not always true and the case is essentially dependent on the complexity of the dynamics of the process, since in case the system dynamics is not suitable to be studied by AR modelling, the excessive model order becomes inevitable.

Through higher sampling rate is desirable for accurate measurements, there are several limitations stemming from computational complications. Among these mention may be made to computer memory requirements for data storage, computer time for the completion of the analysis, facilities for fast sampling and fast data acquisition.

5. CONCLUSIONS

Theoretical and experimental study of MAR modelling for system identification through optimum signal processing is described. Due to the interdependence of the model parameters, in practice care must be exercised to establish favourable modelling conditions suitable for the particular application in hand.

Experimental investigations are carried out by means of a digitally simulated physical system with a well-defined system structure so that precise comparison of the experimental results with their known true values are performed.

There involve three outstanding parameters in MAR modelling, namely, model order \( (p) \), sampling period \( (\Delta t) \) and the \( P \) matrix which is assumed to be diagonal provided the noise sources are uncorrelated. The source of non-diagonal character of the \( P \) matrix is identified to be due to the instantaneous response of the physical system. To reduce the strength of this effect which is uncounted - and therefore undesirable - in MAR modelling, the sampling
rate has to be increased. As far as MAR modelling is considered, sampling rate can be increased as long as the computational facilities as hardware and software permit and some precautions are taken care of e.g. sample-length (n).

For optimum model order the well-known AIC is found to be too restrictive in evaluating the significance of the reduction in the residual variance for univariate analysis and correspondingly the determinant of the $\hat{\Sigma}$ matrix for the multivariate analysis. This assessment is based on the criterion known as F-test in the statistical terminology. From this standpoint, hypothesis testing is found not to be overlooked totally and this is considered in the "penalty term" in the AIC. By doing so, it is observed that the AIC can be used in a less restrictive way to avoid the adoption of excessively high model order in the analysis.

Validity of NCR analysis can be verified by means of cumulative NCR analysis as the latter quantity has to amount to unity. Any deviation from unity would imply eventual false correlations among the noise sources involved.

By the help of the information gained through the studies carried out, conditions to be taken care of for optimum signal processing in AR modelling are determined. Although the parameters involved are related among themselves and they have to be given different values suitable for the particular application in hand, some criteria, namely memory-time and sample length-time play an essential role in AR modelling and they are found to be applicable to each individual case commonly for the establishment of the optimality based on minimum residual variance.

Application of AR modelling technique for system estimation provides several advantages such as possibility for the cause and effect analysis and system dynamics analysis with its noise structure. On the other hand, the technique needs careful provision of the model parameters for reliable and accurate measurements. The parameters are strongly dependent on the nature of signals and the system dynamics under investigation. The dependence is stronger than that of any other conventional analysis methods such as FFT for instance, so that optimal modelling by MAR becomes relatively complicated and therefore requires thorough understanding of the modelling procedure.

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STUDY ON THE GOODNESS OF SYSTEM IDENTIFICATION USING MULTIVARIATE AR MODELING

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ABSTRACT

In order to evaluate the goodness of system identification using a MAR modeling method, simulation study was performed using a hybrid computer. The simulated systems are low order feedback systems with independent Gaussian white noise sources. The test of system identification using MAR modeling method was performed for this simulation model. The results of the simulation study show that the MAR model gives correct estimates of power spectra of the output signals and gain factors of the transfer function between output signals. However, it will not necessarily give correct ones for the estimation of the cut-off frequencies of transfer functions for all sub-systems. The examples in which MAR modeling is applied to real reactor noise data to estimate transfer functions of the control system and the noise sources are also discussed.

KEYWORDS

Multivariate autoregressive model; System identification;

INTRODUCTION

Recently multivariate autoregressive (MAR) modeling technique has become popular in the field of reactor noise analysis as a powerful method for reactor dynamics analysis and diagnosis. For estimating transfer functions and noise sources in a system, however, this method still has many ambiguous points concerning the goodness of the results obtained. In fact, we often experience that the results obtained from MAR modeling differ from our physical understanding. Furthermore, it is well known that the approximation of transfer functions of continuous system by the discrete AR expressions become poor when the AR model order is low.

In spite of such problems, the test of the MAR model fitting for system identification has seldom been performed because the transfer functions of the sub-systems in a feedback system and the system noise signals in a real plant cannot be directly measured and confirmed by usual methods. The objective of the present paper is to study these points through practical MAR model fitting using computer simulated and real reactor noise data. In order to evaluate the goodness of system identification using a MAR modeling method, simulation study was performed using a hybrid computer. The simulated systems are low order feedback systems and are built under the conditions which are required for a MAR model to hold, i.e.,

- each variable has an inherent noise source,
- each noise source is generated from an independent Gaussian white noise source.

Continuous system models are used to make the evaluation reasonable from the view point of frequency range restriction induced by digitising the output signals and the approximation of transfer functions of continuous systems by the corresponding discrete AR expressions.

SYSTEM IDENTIFICATION USING MAR MODEL

MAR Model and Impulse Response Function

Let us consider a 2-dimensional linear feedback system consisting of the sub-systems G12 and G21 as shown in Fig.1. It is assumed that both noise sources u1 and u2 corresponding to the observation variables x1 and x2, respectively, exist in the system. Denoting the impulse response functions of each sub-system as g12(m) and g21(m), both observation signals are expressed as follow;

\[ P_{x-\mathbf{n}} = \sum_{m=0}^{\infty} g_{12}(m) \cdot u_{1,u2}(m) + g_{21}(m) \cdot u_{2,u1}(m) \]
\[ x_i(t) = \sum_{m=1}^{M} g_{ij}(m)x_{j-i}(t-m) + u_i(t) \quad (i, j = 1, 2). \quad (1) \]

Now, consider a problem of estimating \( \{g_{ij}(m); m=1,M\} \) from the observation data \( \{x_i(t); t=-M+1,-M+2, \ldots,N\} \). If \( u_i \) is a Gaussian white noise which is independent from other \( u_j \) \((j \neq i)\), the \( g_{ij} \) can be directly estimated by applying the least squares method to Eq. (1). Otherwise, the estimated \( g_{ij} \) will have a bias error. In order to solve this problem, the following whitened process is introduced.

\[ u_i(t) = \sum_{k=1}^{K} h_{ii}(k)u_i(t-k) + n_i(t) \quad \quad \quad (2) \]

By substituting Eq. (2) into Eq. (1), a system expression which contains the impulse responses \( g_{12}, g_{21}, h_{11} \) and \( h_{22} \) and the white noise sources \( n_1 \) and \( n_2 \) is obtained. This expression is equivalent to the AR model. J-dimensional AR model is defined as follows;

\[ x_i(t) = \sum_{j=1}^{J} \sum_{m=1}^{M} a_{ij}(m)x_{j-i}(t-m) + n_i(t) \quad (i=1, \ldots, J) \quad (3a) \]

where \( a_{ij}(m) \) is an AR coefficient, \( n_i(t) \) a Gaussian white noise and \( M \) the model order.

In the matrix expression, Eq. (3a) can be rewritten

\[ X(t) = \sum_{m=1}^{M} A(m)X(t-m) + N(t) \quad (3b) \]

where

\[ X(t)=\begin{bmatrix} x_1(t) \\ \vdots \\ x_J(t) \end{bmatrix}, \quad A(m)=\begin{bmatrix} a_{11}(m) & a_{12}(m) & \cdots & a_{1J}(m) \\ a_{21}(m) & a_{22}(m) & \cdots & a_{2J}(m) \\ \vdots & \vdots & \ddots & \vdots \\ a_{J1}(m) & a_{J2}(m) & \cdots & a_{JJ}(m) \end{bmatrix}, \quad N(t)=\begin{bmatrix} n_1(t) \\ \vdots \\ n_J(t) \end{bmatrix}. \]

Fig.2(a) shows the flow diagram of MAR model for \( J=2 \). The relationships between the AR coefficients and the impulse response functions are as follows;

\[ g_{ij}(m) = a_{ij}(m) + \sum_{k=1}^{K} h_{ii}(k)g_{ij}(m-k) \quad (m=1, \ldots, M) \quad (4) \]

\[ h_{ii}(m) = a_{ii}(m) \quad (m=1, \ldots, K) \]

\[ = 0 \quad (m=0 \text{ or } K+1, \ldots). \]

Therefore, if \( n_1 \) and \( n_2 \) are independent from each other, the \( g_{12} \) and \( g_{21} \) in Fig.1 can be estimated by applying the least squares method to Eq. (3) even when \( n_1 \) and \( n_2 \) are not white noises.

From the above discussions, we can notice the followings;

- A diagonal element \( [a_{ii}(m)] \) of the AR coefficient matrix gives an impulse response function \( [h_{ii}(m)] \) of the whitened process. This is a backward expression of the impulse response \( [g_{ii}(m)] \) which describes a relationship between the input \( n_i(t) \) and the output \( u_i(t) \).

- An off-diagonal element \( [a_{ij}(m)] \) gives the impulse response \( [g_{ij}(m)] \) by convolution with the whitened process \( [h_{ii}(m)] \).

Estimation of Frequency Response Function

The frequency response function of each sub-system is obtained from Fourier transformed AR coefficients as follows;

\[ G_{ii}(f) = \frac{1}{1 - H_{ii}(f)} = \frac{1}{1 - a_{ii}(f) - a_{ii}(f)} \quad (5a) \]

\[ G_{ij}(f) = G_{ij}(f)a_{ij}(f) = \frac{1}{1 - a_{ij}(f)} \quad (5b) \]

Using these expressions, we can rewrite the signal flow diagram in Fig.1 and Fig.2(a) as Fig.2(b). The frequency response functions of the closed system of \( x_i \) from the inputs \( u_j \) \((i=1, \ldots, J)\) are described by \( G_{ij}(f) \) as follows;

\[ H(f) = (I - G(f))^{-1} \quad (6) \]

where \( I \) is an unit matrix and the matrices \( G(f) \) and \( H(f) \) are defined as

\[ G(f) = \begin{bmatrix} 0 & G_{12} & \cdots & G_{1J} \\ G_{21} & 0 & \cdots & G_{2J} \\ \vdots & \vdots & \ddots & \vdots \\ G_{J1} & G_{J2} & \cdots & 0 \end{bmatrix}, \quad H(f) = \begin{bmatrix} H_{11} & H_{12} & \cdots & H_{1J} \\ H_{21} & H_{22} & \cdots & H_{2J} \\ \vdots & \vdots & \ddots & \vdots \\ H_{J1} & H_{J2} & \cdots & H_{JJ} \end{bmatrix}. \]

Spectral Estimations of Noise Sources and Scores

The spectrum of the system noise \( n_i \) in Fig.2(a) can be written using a white noise source spectrum
System identification using MAR

\[ \text{Pnii}(f) = |\text{Gii}(f)|^2 \text{Pnii}(f) \]  
(7a)

where \( \text{Pnii}(f) = \sigma^2 i1dt \), \( \sigma^2 \) a variance of the white noise source and \( dt \) a sampling interval. If \( \text{nii} \) is not independent from other noise sources \( \text{nij} \), the cross spectrum between \( \text{nii} \) and \( \text{nij} \) exist and Eq. (7a) is described as

\[ \text{Pnii}(f) = |\text{Gii}(f)|^2 \sum_{j=1}^{J} \text{Pnij}(f) \]  
(7b)

where \( \text{Pnij}(f) = \sigma^2 i1dt \).

A contribution of power spectrum of the system noise source \( uj(t) \) to the signal \( xi(t) \) is written using the \( Hij(f) \) of the closed loop system as

\[ Qxuj(f) = |Hij(f)|^2 Pujj(f) \]  
(8)

The power spectrum \( \text{Pxi}(f) \) of a signal \( xi(t) \) which is defined as an output signal of the closed system with input signals \( u1, u2, \ldots \) is written using \( Qxuj(f) \) as

\[ \text{Pxi}(f) = \sum_{j=1}^{J} Qxuj(f) \]  
(9)

The noise power contribution ratio is defined by

\[ \sqrt{xij(f)} = \frac{Qxuj(f)}{\text{Pxi}(f)} \]  
(10)

ANALOG SIMULATION OF MAR MODEL

In order to evaluate the goodness of the system identification using MAR modeling method through practical computation, the simulation data was made using the analog part of a hybrid computer. The use of analog simulated data enables to perform more practical evaluation because the analog simulator can express a transfer function of a continuous system, which requires filtering and digitizing by ADC prior to numerical calculations.

Simulation Model

The simulated model was based on a 2-dimensional feedback system and was added with the coloured process which is characteristic of an AR model. A block diagram of the simulated model is shown in Fig.3. The model consists of the observation variables \( x1 \) and \( x2 \), the white noise sources \( n1 \) and \( n2 \) which are independent from each other, the subsystems \( G11 \) and \( G22 \) as the coloured processes for white noise signals and the subsystems \( G12 \) and \( G21 \) between \( x1 \) and \( x2 \).

White Noise Sources

As the white noise sources \( n1 \) and \( n2 \), two independent noise generators were used. The output level and the frequency range of each noise source are -49dB/Hz between DC and 150 Hz at -3 dB for \( n1 \) and -50 dB/Hz between DC to 700 Hz for \( n2 \). The distribution of each noise source is approximately Gaussian. The coherence function between \( n1 \) and \( n2 \) was less than 0.04 for the whole frequency range.

Transfer Functions of the Sub-systems

The transfer function of each sub-system except for \( G12 \) was set up as a 1-st order time lag system and that of \( G12 \) was a 2-nd order dumping system with a dumping factor 1. Each transfer function is expressed as follows;

\[ \text{Gij}(s) = \frac{K_{ij}}{1 + \omega^2 \frac{\text{Ti}ij}{s}} \quad \text{or} \quad \text{Gij}(s) = \frac{K_{ij} \omega^2 \text{Ti}ij}{s^2 + 2 \omega \frac{\text{Ti}ij}{s} + \omega^2 \frac{\text{Ti}ij}{s}} \]  
(11)

where \( \omega \text{Ti}ij \), \( \omega \text{Ti}ij^2 \), and \( \text{Ti}ij \) are the frequencies and time constants.

Models and Set-up Parameters

Three models with slightly different parameters were used in this simulation study. The model parameters are the gain \( K_i \) and the cut-off frequency \( f _ ij \) of the transfer function \( Gij \). The set up parameters for each model are shown in Table 1. Only four values, 10, 15, 20 and 30 Hz, of the cut-off frequencies were used in order to simplify the simulation, and were set with the following relationships; \( f 11 < f 12 \) and \( f 22 > f 21 \) (model-I), \( f 11 < f 12 \) and \( f 22 < f 21 \) (model-II), \( f 11 > f 12 \) and \( f 22 > f 21 \) (model-III). The gains were set to nearly 4.0 for all cases except for \( G12 \) in the model-III. The loop gains were obtained as about 33 dB (model-I and II) and 10 dB (model-III).

Data Sampling

Six sampled data sets were made for the fitting test of the MAR model. The sampled variables were output signals \( x1 \) and \( x2 \). The sampled data of the case-I, II, III and VI were made from the model-
I, the case-IV from the model-II and the case-V from the model-III. The sampling conditions for each case are shown in Table 2. For the purpose to clarify the effect of data sampling conditions on the fitting, the case-I, II, III and VI were made with different conditions. The different points are as follows. The anti-aliasing filter was not used for the case-I but used for others. The sampling intervals were different among case-I, II and III. The number of data samples was different between the case-VI and others.

IDENTIFICATION TEST AND ITS RESULTS

The MAR model fitting was performed by first estimating the correlation matrices and then solving the multivariate Yule-Walker equation using the Levinson recursion algorithm. The model order was determined by the maximum AIC evaluation (MAICB). The orders of the fitted model for the case-I to VI were 1, 2, 5, 5, 33 and 50, respectively.

Residual Covariance and White Noise Source

A diagonal element \((i,i)\) of the residual covariance matrix in the MAR model means a variance of the \(i\)-th white noise source in physical system and an off-diagonal element \((i,j)\) means a correlation between the \(i\)-th and the \(j\)-th white noise sources. If the \((i,j)\)-element is 0, then the \(i\)-th and \(j\)-th sources are independent from each other.

The covariances obtained in each fitting are shown in Table 3. The obtained off-diagonal terms show fairly high values between 0.29 to 0.59, except for the case-III, and it seems that these results indicate a strong correlation between the white noise sources \(n_1\) and \(n_2\). However, the reason should be referred to other factors because the white noise sources used are independent from each other at least in the analog form. One of the possible cause is the number of data samples. By comparison of these results for the case-II \((N=4096)\) and the case-VI \((N=101376)\), it is found, however, that the correlated value is little improved.

The diagonal terms were smaller than the true values of the variances. Table 4 shows the spectral values \(P_{n1}\) and \(P_{n2}\) of the white noise sources \(n_1\) and \(n_2\) calculated from each variance and also shows its estimation error. The errors of \(P_{n1}\) and \(P_{n2}\) lie randomly between -24.3 and +2.2 dB and between -17.1 and +5.7 dB, respectively.

Table 5 shows the spectral values \(P_{n1}(0)\) and \(P_{n2}(0)\) of the system noise sources \(u_1\) and \(u_2\) calculated from \(P_{n1}\) and \(P_{n2}\) and the gain of the coloured processes \(G1(0)\) and \(G2(0)\) described later. In this simulation, \(P_{n1}\) and \(P_{n2}\) at 0 Hz theoretically agree with \(P_{n1}\) and \(P_{n2}\), respectively. The obtained spectral values except for the case-I agree with \(P_{n1}\) and \(P_{n2}\) within 3 dB for \(P_{n1}\) and within 5.4dB for \(P_{n2}\).

From these results, the following points were found;

- the diagonal terms of the residual covariance matrix is not equivalent to the variances of the system white noise source and these terms will have physical meanings only when combined with the gain of the coloured process.
- The aliasing components caused by data sampling made it difficult to estimate the noise sources. (see Case-I)

Spectral Estimations of Signals \(x_1\) and \(x_2\)

With regard to the spectral estimation of the observation signals, it is well known that the AR spectrum agrees with those by other methods. In this paper, the comparison of AR estimates with those measured using a FFT signal analyzer was made for a case of the fitting to the model-I data. Figs.4 and 5 show APSDs of signals \(x_1\) and \(x_2\), respectively. These APSDs were estimated using 512 data samples with an averaging number of 1000 for five partly overlapping frequency bands of the data samples and the frequency bandwidths 2, 12.5, 64, 256 and 800 Hz. Fig.6 shows the relative estimation errors of AR spectrum measured at arbitrary frequency points where the FFT spectrum is put 0 dB as a reference value. The number of frequency points in the Fourier transform of AR coefficients was 500 in this paper. The abscissa is the normalized frequency \((f/f_{max})\).

It shows that the results for the case-II and VI agree with the FFT spectrum within 1.5 dB. Since both case-II and VI only differ in the data size which are 4096 and 101376, respectively, it suggests that 4096 is a sufficient number for this spectral estimation. The case-I has a large error which is 2 to 3 dB in the lower frequency and 5 to 10 dB in the higher frequency. This is considered to be due to poor resolutions of both A/D converters of the FFT analyzer and the hybrid computer used because the simulated signal has a large power in the lower frequency range and each input voltage range of the A/D converter was fitted to this. From these results, it is confirmed that MAR spectrum can be estimated with a sufficient accuracy if proper signal preprocessing is performed.

Coloured Processes \(G1\) and \(G2\)

Both transfer functions \(G1\) and \(G2\) of coloured processes in the simulated system are 1st order lag elements with gains 1.0 (0 dB) but different cut-off frequencies for different simulated model.

Table 6 shows the estimated gains and cut-off frequencies of the transfer functions \(G1\) and \(G2\) which were obtained using a procedure for fitting a transfer function to the estimates of
frequency response function.
The gain has a certain value and it compensates the residual covariance which is estimated too small as described in the previous section. In practice, the magnitude of the noise source spectrum in the lower frequency range well agreed with that of the true spectrum.

The forms of the functions of the estimated coloured processes were approximated by the rational expressions whose denominator order MD and numerator order MN were 1 or 2 although for the setup transfer function MD=1 and MN=0. This means that the estimation of the function forms was unsuccessful.

The cut-off frequencies shown in Table 6 are the lowest ones of the obtained transfer function. Most of results differ from the true values. In order to evaluate the estimation errors of the cut-off frequency, the ratios of the estimated versus the setup frequencies \( f/f_{0} \) were calculated in dB unit and are shown in Table 6. The errors differ from case to case. The case-II and VI which indicated the best results in the spectral estimation for the model-I show that the errors for G11 are -2.7 and -0.5 dB but for G22 are -7.5 and -13.4 dB, respectively. From these and also the case-V results, it seems that the error for G11 is smaller than for G22, which suggests that the error depends on the calculation order of G11 and G22 in fitting procedure. On the other hand, the errors for the case-III and IV for which too small sampling intervals were used, show all positive values but all negative values for other cases. It suggests that the error also depends on the sampling intervals.

Transfer Functions G12 and G21
In this simulation, the transfer functions G12 and G21 between the signal x1 and x2 were 2-nd and 1-st order time lag systems, respectively. The setup value of the gain was 1.0 (0 dB) for all cases except for the G12 of the model-III whose gain was 0.7 (-3 dB). The cut-off frequency of G12 was fixed to 20 Hz and that of G21 was 15 Hz or 30 Hz.

The estimation results of these parameters are shown in Table 6. The gain value was estimated correctly within 1 dB except for the case-I which contained the aliasing components. The parameters of the function form, the cut-off frequency and the damping factor of G12 were not estimated correctly the same as the estimation for G11 and G22 because these functions are calculated by Eq. (5b) using estimates of the coloured processes, and the errors depend on it. Therefore, if the estimation of the coloured process fails, the estimation of the transfer function between the observation variables also fails.

Transfer Functions H12 and H21 of the Overall System
The transfer function of the overall system between a system noise source as input and an observation signal as output is calculated from the transfer functions of the sub-systems and is used for noise power contribution ratio analysis. In a practical system, this transfer function cannot be estimated directly because the system noise sources are internal variables.

In the simulation, this transfer function was directly measured by the FFT analyser using a system noise source signal and was used as a reference function for comparison. Fig.7 shows an example of H11 of the model-I. It is found that although the estimates have some frequency bias errors, the estimates of the MAR fitting keep an overall characteristics of the reference transfer function and that both gain and phase in the lower frequency range particularly agreed in each case.

Noise Power Contribution Ratio
The noise power contribution (NPC) ratio of the system noise source \( x_{i} \) to the observation signal \( x_{j} \) is defined by Eq.(10). On the other hand, the coherence function between \( x_{i} \) and \( u_{j} \) is calculated as follows;

\[
\Gamma_{x_{i}u_{j}}(\tau) = \frac{|P_{x_{i}u_{j}}(\tau)|^2}{P_{x_{i}}(\tau)P_{u_{j}}(\tau)} = \frac{|H_{i,j}(\tau)P_{u_{j}}(\tau)|^2}{P_{u_{j}}(\tau)P_{x_{i}}(\tau)} = \Gamma_{x_{i}j}(\tau).
\]  (12)

Therefore, the NPC ratios for the simulation model can be directly measured by the FFT analyser. Fig.8 shows the estimated NPC ratios for the model-I. It is indicated that the obtained NPC ratio has a certain frequency bias error because the estimation error depends on the accuracy of \( H_{i,j}(\tau) \) but that the estimates keep the overall characteristics of the reference NPC ratio.

Time Constants Estimation by Step Response
Estimation of the time constant of the system using a step response function is performed by detecting the time for the step response to reach 63.2\% of the final steady state. This method assumes that the time constant defined for a 1-st order time lag system gives an approximation of the largest time constant of the system.

The results of the time constant estimation for each sub-system and the closed system are shown in Table 7. The maximum calculation points in the time axis is 500. From the results, it is found that the case-III, IV and V give relatively correct estimates for the sub-system G12 and G21 and that the case-II, VI and V also give correct estimates for the closed loop systems H12 and H21. These results can be understood as follows. The resolution for a time axis of the step response function depends on the sampling interval. On the other hand, the cut-off frequencies of the sub-systems used in the simulation are 10, 15, 20 and 30 Hz and the corresponding time constants are
15.9, 10.6, 7.96 and 5.31 msec, respectively. Therefore, the case-I, II and VI in which large sampling intervals as 25 or 12.5 msec are used cannot expect satisfactory results. Furthermore, because the time constants of the closed loop system of the model-I and II are too large as 1.6 sec, the case-III and IV in which too small sampling intervals as 1.25 msec is used cannot expect satisfactory results when a maximum calculation point is set to a small number.

From these results, it is found that the estimation of a time constant of the sub-system or the closed loop system gives a relatively correct value if the estimated system is of lower order and if the sampling interval and the maximum calculation point in the time axis are properly selected.

APPLICATION TO NSRR NOISE ANALYSIS

In this section, an example of the application of the system identification using MAR modeling to the noise data from a pulsed reactor NSRR (Nuclear Safety Research Reactor) is shown. The reactor NSRR has shown that the reactor power fluctuates when it is operated at around 300 kW, causing reactor scram in certain cases. To diagnose the cause of abnormal fluctuations of the reactor power, the reactor noise experimental analysis was performed. In the experiment, it was sometimes observed that the reactor power oscillated significantly with a frequency of 0.3 Hz. It was concluded from this diagnostic analysis that the abnormal fluctuation of the reactor power was due dominantly to the automatic control system which shows poor dynamic performance when relatively large disturbances are applied by the operation of the N16 diffuser system.

The automatic control system constitutes a feedback system from the reactor power to the reactivity. Therefore the dynamics analysis of the automatic control system requires a method which is able to treat a feedback system. On the other hand, if the variables treated include the output signals which are observed at a point between two serially connected open loop transfer functions, the MAR model fitting becomes difficult because the MAR modeling has a feature that the model is made by considering all signal paths. Such an example is the internal variables of the automatic reactor control system. The simplest way for fitting in such a case is to select the variables which have physically paths to other variables.

The neutron signal and the bank indicator signal which are internal variables of the control system as shown in Fig.9, were selected here and the model fitting was performed using the data with a sample size of 30080. The MAICE order was obtained as 50 and the off-diagonal term of the residual covariance was 0.16. The estimates of the frequency response function H11 of the closed loop system is shown in Fig.10. The H11 is the frequency response function of the reactor power fluctuation signal from its additive noise, i.e. the disturbance component to the control system. The disturbed components occurs in the reactor dynamics. However it disappears as the output signal of a transfer function of the reactor dynamics G(s) in Fig.9 is treated as the additive noise. The results show that the gain has a peak of 3 dB at 0.3 Hz. This means that if some disturbance occurs with around 0.3 Hz, then it is amplified by the feedback loop and disturbs the control system. In fact, this frequency agreed with the dominant frequency of the observed oscillation. This analysis is not so accurate but the result is reliable enough because the characteristics of interest lie in the lower frequency and the number of data samples is large enough as described in the previous section.

CONCLUDING REMARKS

In order to evaluate the goodness of system identification using MAR modeling method, 2-dimensional feedback systems simulating the MAR model structure were built on a hybrid computer and the practical identification test was performed using the samples of the simulated data. The following results were obtained through the fitting test.

- The signal power spectrum can be estimated correctly.
- The residual covariances of a fitted MAR model are smaller than those of actual white noise sources and the gain values of the coloured process compensate these errors.
- Estimation error for the coloured process is large. The gain level of the obtained frequency response function is relatively correct in the low frequency range but the factors determining characteristics in the higher frequency range such the function type and cut-off frequency are fitted poorly because the noise source levels are generally small and also because the MAR fitting algorithm has such a structure that fitting errors are accumulated in this range.
- Estimations of the system noise source spectrum and the frequency response functions of the observation signals are also bad in the higher frequency range because these functions are calculated using the estimates of the coloured processes and the errors depend on it.
- As for the frequency response function of the closed loop system between the signals and the noise sources, the estimates keeps an overall characteristics although the estimates have some bias errors in the lower frequency range and large errors in the higher frequency range.
- The estimates of the noise power contribution ratio also keep the overall characteristics in lower frequency.
- The estimation of the time constant of the sub-system or the closed loop system gives a relatively correct value if the estimated system is of lower order and if the sampling interval and the maximum calculation point in time axis are properly selected.
REFERENCES


Table 1. Parameters of the simulated model

<table>
<thead>
<tr>
<th>Model</th>
<th>System Order</th>
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<th>fc (Hz)</th>
<th>DF</th>
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<td>G11</td>
<td>1</td>
<td>0.9995</td>
<td>10Hz</td>
</tr>
<tr>
<td></td>
<td>G12</td>
<td>2</td>
<td>0.9995</td>
<td>20Hz 1.0</td>
</tr>
<tr>
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<td>G21</td>
<td>1</td>
<td>0.9995</td>
<td>15Hz</td>
</tr>
<tr>
<td></td>
<td>G22</td>
<td>1</td>
<td>0.9995</td>
<td>30Hz</td>
</tr>
<tr>
<td>II</td>
<td>G11</td>
<td>1</td>
<td>0.9995</td>
<td>10Hz</td>
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<tr>
<td></td>
<td>G12</td>
<td>2</td>
<td>0.9995</td>
<td>20Hz 1.0</td>
</tr>
<tr>
<td></td>
<td>G21</td>
<td>1</td>
<td>0.9995</td>
<td>30Hz</td>
</tr>
<tr>
<td></td>
<td>G22</td>
<td>1</td>
<td>0.9995</td>
<td>15Hz</td>
</tr>
<tr>
<td>III</td>
<td>G11</td>
<td>1</td>
<td>0.9995</td>
<td>20Hz 2.0</td>
</tr>
<tr>
<td></td>
<td>G12</td>
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<td>0.7000</td>
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<td>0.9995</td>
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<td></td>
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Table 2. Sampling conditions

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<th>Model</th>
<th>Gain (dB)</th>
<th>LFF (msec)</th>
<th>dt (sec)</th>
<th>freq-range (Hz)</th>
<th>(Points)</th>
<th>(Ks)</th>
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<td>4096</td>
<td>0.04-20</td>
<td></td>
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<tr>
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<td>I</td>
<td>20</td>
<td>40</td>
<td>12.5</td>
<td>51.2</td>
<td>4096</td>
<td>0.08-40</td>
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<tr>
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<tr>
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Table 3. Residual Covariances

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<th>22</th>
<th>12 (Normalized)</th>
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<td>6.90E-4</td>
<td>2.51E-4 (0.56)</td>
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<td>III</td>
<td>5</td>
<td>3.70E-5</td>
<td>4.33E-4</td>
<td>1.07E-5 (0.08)</td>
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<td>IV</td>
<td>5</td>
<td>4.96E-5</td>
<td>1.57E-4</td>
<td>2.87E-5 (0.33)</td>
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<tr>
<td>V</td>
<td>35</td>
<td>8.06E-5</td>
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<td>1.86E-4</td>
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<td>50</td>
<td>3.86E-4</td>
<td>4.70E-4</td>
<td>2.23E-4 (0.52)</td>
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<td>5</td>
<td>3.86E-4</td>
<td>4.74E-4</td>
<td>2.24E-4 (0.52)</td>
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Table 4. Magnitudes (dB) of white noise source n1 and n2

<table>
<thead>
<tr>
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<th>Error</th>
<th>Pn22</th>
<th>Error</th>
</tr>
</thead>
<tbody>
<tr>
<td>I</td>
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<td>-44.3</td>
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<td>II</td>
<td>-54.4</td>
<td>-5.4</td>
<td>-50.6</td>
<td>-0.6</td>
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<td>III</td>
<td>-73.3</td>
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<td>-12.7</td>
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<td>IV</td>
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<td>-67.1</td>
<td>-17.1</td>
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<td>V</td>
<td>-63.9</td>
<td>-14.9</td>
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<td>-11.0</td>
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<tr>
<td>VI</td>
<td>-63.3</td>
<td>-14.3</td>
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Table 5. Magnitudes (dB) of coloured noise sources u1 and u2

<table>
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<th>Error</th>
<th>Pn22</th>
<th>Error</th>
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<tbody>
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<td>-41.4</td>
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<td>II</td>
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<td>-2.1</td>
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<td>+0.2</td>
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<tr>
<td>V</td>
<td>-45.9</td>
<td>+2.1</td>
<td>-49.0</td>
<td>+1.0</td>
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<td>VI</td>
<td>-47.2</td>
<td>+1.7</td>
<td>-45.3</td>
<td>+5.4</td>
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Fig. 1 Block diagram of a 2-dimensional linear feedback system.
Table 6. Estimated gains and cut-off frequencies using frequency response function

<table>
<thead>
<tr>
<th>Tans. Model Set-Up</th>
<th>MAR Estimation</th>
<th>Error</th>
<th>(f/fe)</th>
</tr>
</thead>
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<td>fc</td>
<td>CASE M</td>
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<tr>
<td></td>
<td>(dB)</td>
<td>(Hz)</td>
<td>(dB)</td>
</tr>
<tr>
<td>G11</td>
<td>I 0</td>
<td>10</td>
<td>I 1</td>
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</tr>
<tr>
<td>G22</td>
<td>I 0</td>
<td>30</td>
<td>I 1</td>
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<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>G21</td>
<td>I 0</td>
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<td>I 1</td>
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<tr>
<td>G12</td>
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Table 7. Estimated time constants using step response function

<table>
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<th>MAR Estimation</th>
<th>Error</th>
<th>(f/fe)</th>
</tr>
</thead>
<tbody>
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<td>(sec)</td>
<td>(Hz)</td>
<td>(sec)</td>
</tr>
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Fig. 2 Block diagram of a 2-dimensional AR model

Fig. 3 Block diagram of the simulated feedback system.
Fig. 4 APSD of signal $x_1$ (the model-I). Fig. 5 APSD of signal $x_2$ (the model-I).

Fig. 6 Estimation errors of AR spectra (the model-I).

Fig. 7 Frequency response function $H_{11}$ (the model-I).
Fig. 8 Noise power contribution ratio (the model-I).

Fig. 9 Block diagram of the automatic control system of NSRR.

Fig. 10 Frequency response function of the closed loop system of reactor power signal from its additive noise source.
A METHOD OF NONSTATIONARY NOISE ANALYSIS USING INSTANTANEOUS AR SPECTRUM AND ITS APPLICATION TO BORSSELE REACTOR NOISE ANALYSIS

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ABSTRACT
This paper describes the test and demonstration of a method of nonstationary noise analysis using frequency-time spectrum based on instantaneous AR spectra. The frequency-time spectra were estimated by dividing the sample record into a series of very short subrecords which can be considered to be locally stationary and by calculating instantaneous spectrum for each subrecord using univariate AR modeling. This method was applied to the analysis of Borssele reactor noise data measured during shut-down operation, which shows clear dependence of several signals on the coolant temperature.

KEYWORDS
Non-stationary analysis; Instantaneous AR spectrum; Frequency-time spectrum;

INTRODUCTION
Nonstationary phenomena in reactor noise observed during transient operation such as reactor start-up and shut-down contain very important information sources which are very useful for diagnosis of nuclear power plants. In the field of reactor noise analysis, however, nonstationary noise analysis has seldom been performed because there is no effective analysis method established for nonstationary noise data and also because noise data measurement during transient reactor operation is much more difficult than during steady state operation.

The analysis of non-stationary data requires a method which is able to deal with the characteristics of its non-stationarity. With regard to the non-stationary data whose characteristics change very slowly, the frequency-time spectrum using normal FFT spectra has been applied for the analysis when the data is considered to be locally stationary. However, the resolution in the time axis by this method decreases for the case of small sample data. An example has also been used in which the least squares AR method applicable to the analysis of a small size sample data was used for the analysis of rapidly changing transient data.

In the method of non-stationary noise analysis described in the following sections, the frequency-time spectrum based on instantaneous spectra is applied. For estimating the instantaneous spectra, AR modeling technique based on least squares method is used because it is possible to analyze a relatively small number of data samples.

INSTANTANEOUS SPECTRAL ANALYSIS OF NON-STATIONARY DATA
Instantaneous Mean Squares
For a stationary signal \( x(t) \), the mean squares value is defined as the limit of the time average for an infinite observation time as follows;
\[
\Psi_x^2 = \lim_{T \to \infty} \frac{1}{T} \int_0^T x(t)^2 \, dt.
\]
For a very short time span \((t, \Delta T)\), Eq. (1) is replaced by

707
\[
\Psi x^2(t,\Delta T) = \frac{1}{\Delta T} \int_0^{\Delta T} x(t)^2 \, dt \tag{2}
\]
which gives an instantaneous mean squares. Both estimates of Eqs.\,(1) and \,(2) have bias errors when the signal is non-stationary. However, Eq.\,(2) gives an unbiased estimate for a limited case where the signal during the time span \((t,\Delta T)\) is locally stationary, although the variance error increases.

**Instantaneous AR model**

An AR model of a stationary process \(x(t)\) is given by

\[
x(t) = \sum_{m=1}^{M} a(m) x(t-m) + n(t). \tag{3}
\]

For a process which is locally stationary in a very short time span \((t,\Delta T)\), the instantaneous AR model for the \(k\)-th time span can be expressed by

\[
x(t) = \sum_{m=1}^{M(k)} a(m,N(k),k) x(t-m) + n(t,k) \tag{4}
\]

where \([a(m),N(k),k] \) are AR coefficients, \(n(t,k)\) a gaussian white noise having a mean 0 and a variance \(\sigma^2(k)\) and \(N(k)\) the order of AR model. Eq.\,(4) is essentially equivalent to a stationary AR model and both coincide when \(\Delta T \to \infty\).

For a time varying AR model which represents systematically some non-stationary characteristics, the white noise term is represented by \(n(t)\) and the coefficients of the time varying AR model becomes different from those in Eq.\,(4). In the present analysis, Eq.\,(4) is used.

**Instantaneous spectrum**

Using the instantaneous AR model, an instantaneous spectrum \(P(f,k)\) for the \(k\)-th time span is obtained by

\[
P(f,k) = \frac{\sigma^2(k)\Delta t}{\left| 1 - a(f,N(k),k) \right|^2} \tag{5}
\]

where \(\Delta t\) is a sampling interval, \(a(f,N(k),k)\) the Fourier transform of \(a(m,N(k),k)\) which is determined by

\[
a(f,N(k),k) = \sum_{m=1}^{M(k)} a(m,N(k),k) \exp(-2\pi ifm\Delta t) \tag{6}
\]

**AR Model Fitting Method**

The AR model fitting is a process of determining the order \(M\), the coefficients \([a(m); m=1,\ldots,M]\) and a variance \(\sigma^2\) characterizing a gaussian white noise using the observed data \([x(t); t=1,\ldots,N]\), which leads to a problem of determining \((M+2)\) unknown parameters.

Four types of AR model fitting algorithms are discussed here for determining each AR parameter from the measured data. They are Yule-Walker, Burg, Marple and Kitagawa-Akaike methods.

**Yule-Walker Method**

A linear equation called Yule-Walker normal equation is derived by multiplying Eq.\,(3) by \(x(t)\) and taking the expectation,

\[
R_M A_M = \Sigma_M \tag{7}
\]

where

\[
R_M = \begin{bmatrix} R(0) & R(-1) & \cdots & R(-M) \\ R(1) & R(0) & \cdots & \vdots \\ \vdots & \vdots & \ddots & \vdots \\ R(M) & \cdots & \cdots & R(0) \end{bmatrix}, \quad A_M = \begin{bmatrix} 1 \\ a(1,1) \\ \vdots \\ a(M,N) \end{bmatrix}, \quad \Sigma_M = \begin{bmatrix} \sigma^2 \\ 0 \\ \vdots \\ 0 \end{bmatrix}.
\]

Eq.\,(7) can also be obtained by determining the coefficients of an AR model, viewed as a linear prediction filter, which minimize the mean squares of prediction error. The \((M+1)\) unknown parameters, consisting of \(M\) AR coefficients and a covariance, are obtained by substituting an estimated correlation function \([R(k); k=0,1,\ldots,M]\) into Eq.\,(7). Using Levinson recursion algorithm, the solutions for higher order models are obtained efficiently once the AR coefficients for \(M=1\) are determined. We will call this procedure as Yule-Walker method for convenience in this paper. The optimal model order \(M\) as the final unknown parameter is determined using the AIC.

Successful fitting by this method is dependent on the accuracy of estimated correlation function. For a small number of data samples, the correlation function is usually computed using biased lag estimates in order to guarantee a positive-definite correlation matrix. However, the fitting error of the AR model becomes large if the sample size is small.
Burg Method
This method is an improved one in which a constrained least squares method is applied to determine AR coefficients in order to suppress the bias error of correlation estimates, a disadvantage of the Yule-Walker method. The final AR coefficient \( a(M,M) \), i.e., a reflection coefficient, is directly estimated from data samples by minimizing the sum of the forward and backward linear prediction energies. The Burg formula derived theoretically is used in this calculation. If a variance \( R(0) \) of data samples and \( a(1,1) \) are known, then \( R(1) \) and \( a^2 \) are obtained from Eq. (7). \( R(k); k=1, \ldots, M \) and AR parameters except for \( a(M,M) \) in higher order models are also obtained using Levinson equation.

Successful fitting by this method depends on the estimation accuracy of \( R(0) \). Therefore, the fitting error by this method is smaller than by the Yule-Walker method.

Marple Method
Marple method is an advanced algorithm which expands the idea of the Burg method. The principal idea is to apply the exact least squares method to determine not only the reflection coefficients but also all other AR coefficients. From this idea, the following equation is derived.

\[
R_N A_M = E_M
\]

where

\[
R_N = \begin{bmatrix} r(0,0) & \cdots & r(0,M) \\ \vdots & \ddots & \vdots \\ r(N,0) & \cdots & r(N,M) \end{bmatrix}, \quad A_M = \begin{bmatrix} 1 \\ a(1) \\ \vdots \\ a(M) \\ 0 \end{bmatrix}, \quad E_M = \begin{bmatrix} e_1 \\ 0 \\ \vdots \\ 0 \end{bmatrix}
\]

\[
r(i,j) = \sum_{t=1}^{N-M} (x(t+N-j)x(t+N-1) + x(t+j)x(t+1)).
\]

Eq. (9) is a function defined as the sum of the forward and backward correlation functions without time-mean operation. The AR parameters are obtained as a solution of Eq. (8). For solving Eq. (8), it is impossible to use the Levinson recursion algorithm because \( R_M \) in Eq. (8) is not a Toeplitz matrix. However, instead of the Levinson algorithm, there exists a recursion algorithm for solution of Eq. (8) which can be computed efficiently though complicated. Furthermore, if initial values for Eq. (9) at \( M=1 \) are obtained, Eq. (9) for higher order models can be calculated efficiently by use of recursive relationship.

Kitagawa-Akaike Method
Kitagawa-Akaike method provides an algorithm which uses the exact least mean squares method as is the case in the Marple method. The AR coefficients are determined by minimizing the sum of either the forward or the backward prediction error. The problem of determining AR coefficients which minimize the prediction error is reduced to solving the following \( (N-M) \) simultaneous linear equations.

\[
X^T A = X^T Y
\]

where

\[
X = \begin{bmatrix} x(M) & x(M-1) & \ldots & x(1) \\ x(M+1) & x(M) & \ldots & x(2) \\ \vdots & \vdots & \ddots & \vdots \\ x(N-1) & x(N-2) & \ldots & x(N-M) \end{bmatrix}, \quad Y = \begin{bmatrix} x(M+1) \\ x(M+2) \\ \vdots \\ x(N) \end{bmatrix}, \quad A = \begin{bmatrix} a(1) \\ a(2) \\ \vdots \\ a(M) \end{bmatrix}
\]

Using the QR decomposition of a rectangular matrix

\[
X = QR
\]

where \( Q \) is a \((N-M)xM\) rectangular matrix consisting of column vectors in the orthonormal system and \( R \) is an upper triangular matrix, Eq. (10) is rewritten as

\[
R A = Q^T Y
\]

which can be solved by only backward substitution. This QR decomposition is realized by use of a Householder transform matrix.

In this method, the solutions are obtained for not only a maximum order \( M_{max} \) but also for other lower orders at one time of computation of \( M_{max} \). However, the fitting by the one time computation is not appropriate for the case of small sample size because the number of data samples used for fitting of lower order models is fixed as \((N-M_{max})\).

**SIMULATION STUDY**

**Tests of Four Fitting Methods**
Four AR methods discussed in the previous section were tested in regard to fitting to a small size
samples\(^5,6\)). The test data were sampled with a sampling interval of 0.061 sec from a sinusoidal analog signal having a frequency of 1.5 Hz and an amplitude of 2.5. About 12 samples are obtained in a period. When the spectrum of the signal contains line-spectral components, the AR fitting which assumes the white noise source generally becomes difficult. However, in the present test, the condition is slightly better than for purely sinusoidal data because some quantization errors are introduced through AD conversion.

Figs. 1 and 2 shows the AR spectra estimated by each method for the modal order \(N=2\) and \(N=3\), respectively. The number of the data samples \(N\) was 6 or 7. The total number of the data spans which were taken from the original data was 99. In this analysis, each data span overlaps for \(N-1\) samples, that is, the length of an updated part is 1.

From these Figures, it was found that the estimated spectral curves were different for different data spans and also for different fitting methods. It can be explained by the following reasons. When the data spans are taken with a length less than the period of the sinusoidal signal, the mean and variance values cannot be estimated exactly because each truncated sinusoidal data apparently has non-stationary characteristics.

The estimation results of the magnitude and the frequency of the original sinusoidal signal are shown in Figs. 3 and 4. Note that the results depend on the period rather than the number of data samples. The results shows that both Marple and Kitagawa-Akaike methods, which are based on the exact least squares methods, are superior to other methods for the estimation of the peak frequency but any method can obtain a satisfactory result for the estimation of the peak height. This is due to the assumption of the stationarity which is required in the AR modeling, and for the estimation of statistical parameters, e.g., a mean and a variance, a sufficient number of sample data which can be considered as stationary is required.

Non-stationary Case

In order to perform estimation tests of the frequency-time spectrum, non-stationary noise data of which stochastic properties change slowly in time was simulated by a system shown in Fig. 5. Three signal sources, a white noise, triangular waves with a fixed base frequency of 8 Hz and a variable base frequency between 15 Hz and 26 Hz, were used. These signals were mixed and then passed through a shaping filter of 2nd order system having a damping factor of 0.05 and a cut-off frequency of 5 Hz. The output signal was sampled with a sampling interval of 0.01 sec. The total number of the data samples was 4096 and the time length was 40.96 sec. Fig. 6 shows the sampled data recordings.

The power spectrum estimated using all the sampled data is shown in Fig. 7. Three remarkable peaks at 5, 8.5 and 26.5 Hz and trapezoidal components between 17 and 26 Hz are found. Fig. 8 shows the result of the small data span analysis applied to the same data. The length of each data span was selected to be 1024 samples, of which 960 samples overlap with the successive span and 64 samples are updated.

The upper graph in Fig. 8 is a frequency-time spectrum and the lower one is the power spectra for each data span. It is found from the upper graph that a line spectral component at 27 Hz is observed during the initial 13 spans, and the form and the location of this component change after 14-the span whose time range is from 8.33 to 18.56 sec.

Fig. 9 shows the results for the case where the length of the data spans was set to be 128 samples. Each data span was taken without overlapping from the original data. Although spectral forms are estimated roughly, three main peak components appear clearly. In particular, it is clearly found that the trapezoidal component which appeared in the previous analysis is an afterimage of a line-spectral component with a shifting frequency. The beginning point of the frequency change is the 14-th span. This span number is the same as in the previous analysis but the time range in this case is narrower and is from 16.65 to 17.92 sec.

The above result indicates the followings:
- When the length of data span as an analysis time window is short in consideration of the change speed of the time dependent components, the original component can be well estimated.
- When the length is long, the magnitude of the estimated spectral component decreases and the spectral distribution spreads if the original signal has a frequency shifting components.

APPLICATION TO BORSSELE REACTOR NOISE

With respect to the pattern changes in the power spectra of the primary loop pressure signals and the axial vibrations of primary coolant pumps in Borssele reactor, the analysis using normal FFT spectrum was already reported\(^7\). The analysis was able to follow well the pattern changes in spite of the use of FFT spectrum which requires a large number of data samples because the pattern changes of these signals were very slow.
In this section, we discuss about Borselle reactor noise analysis using an instantaneous AR spectrum from the view point of non-stationary data analysis.

Data Records
The data were measured during a shut-down operation at the end of 11-th core cycle. Four signals, primary loop pressure (Y01.PO02), coolant temperature of core exit (X053.T01), differential pump pressure (Y01.PO03) and radial vibration of primary pump (Y01.V003) were selected for this analysis.

These data records were divided into 2 files, B11-243 and B11-244, from a reason of limitation of maximum recording time. The recordings of the B11-244 were started after 6 minutes of the end of the B11-243. The DC signal records in each file are shown in Fig.10 (a) and (b). For the B11-243, both coolant temperature and loop pressure are initially constant and decrease after at 21300 sec from 150 to 85 ata and from 262 to 210 C, respectively. The differential pump pressure increases during that time interval from 5.9 to 6.35 ata. The pump vibration signal does not change.

Mean Squares Series
Fig.11 shows the results of the mean squares series analysis applied to the part of each file data for which appears large changes. The analysis of both files was made for the ranges from 16000 to 24992 sec and from 1 to 8192 sec, respectively.

Each mean squares value was estimated for every 2 sec (128 samples). The results shown in Fig.11(a) indicate that the variance component of the loop pressure decreases once after 21300 sec and then decreases again slowly. The mean component of the pump vibration increases after the same time point and the coolant temperature has a super low frequency with a period of 1000 sec.

The results in Fig.11(b) indicate that the variance component of the loop pressure becomes progressively large. The increase of the mean component of pump vibration stops at 4000 sec when the pressure DC signal becomes constant. The period of the super low frequency component appearing in the coolant temperature changes from 1000 to 2000 sec after the time point at 3000 sec.

Frequency-Time Spectrum
The frequency-time spectra are shown in Figs. 12 to 15. The upper graph (a) in each figure is the result for the B11-243 for the time range (15800-25008 sec) and each lower graph (b) is the result for the B11-244 for (5200-14208 sec). Each data span was the first 8 sec (1024 samples) taken from each 200 sec segment of the original data.

For the case (a) of the loop pressure spectra, the remarkable peaks at 1.5, 6.8, 14 and 16.5 Hz are seen in the initial time span. When the DC components described above changes, the frequencies of two peaks at 6.8 and 14 Hz begin to shift towards higher frequencies. In the initial span for the case (b), the peaks at 1.5, 8, 11.5 and 16.5 Hz are observed. The peak at 8 Hz is the result of shifting of the 6.5 Hz peak in the case (a). The peak at 11.5 Hz is from the 9 Hz peak which was not remarkable in (a). The 15.5 Hz of the peak shifted from 14 Hz of (a) is mixed with the large component of the 16.5 Hz. These frequencies begin to shift again at the time point of 5900 sec and stop at 12000 sec. These time points correspond to the beginning and the end points of the decrease of the coolant temperature. It is found that the shifting lines of the peak frequencies are distorted at 9850 sec by the stop of the decrease of the loop pressure.

From Fig.13, it is found that the coolant temperature consists of the colored noise with two sharp peak components at 12 and 25 Hz and that there is no frequency shifting peak. On the other hand, the periodic change of the spectral component at a lower frequency is observed. This indicates a super low frequency oscillation with a period of 1000 sec detected by the mean squares analysis.

The differential pump pressure in Fig.14(a) has remarkable peaks at 11 and 17.5 Hz and a weak peak at 9.6 Hz. When the temperature and pressure begin to change, the peak at 9.6 Hz begins to shift then overlaps with the large peak component at 11 Hz. The peak at 17.5 Hz disappears at the same time. In Fig.14(b), the shifted frequency component which originated from 9.6 Hz does not clearly appear even when the temperature or pressure changes.

The pump vibration in Fig.15(a) has many remarkable peaks. These are at 2.7, 3.8, 7.4, 9.5, 14 and 18 Hz. When the temperature and pressure changes, two peaks at 9.5 and 18 Hz begin to shift towards higher frequencies. In Fig.15(b), a peak at 8 Hz newly appears as a shifted frequency component. These peaks shift in a similar manner as in the loop pressure spectra.

From these results, it is confirmed that the frequency-time spectrum based on instantaneous AR spectra is capable to analyze non-stationary data for a detailed time scale if the change of non-stationary characteristic is slow. The combined use of this spectrum and other time information, i.e., DC signal recordings and instantaneous mean squares series, is very useful for these
analyses.

CONCLUDING REMARKS

A frequency-time spectrum method based on instantaneous AR spectra was applied to the analysis of non-stationary data whose stochastic characteristics changed slowly. Four fitting methods of AR model were tested in order to investigate their applicability for non-stationary data analysis.

From the results, the following points were confirmed:

- Bias errors of estimated peak frequencies are large by both conventional methods of Yule-Walker and Burg method but small by the exact least squares AR methods. They are Marple and Kitagawa-Akaike methods. For the case of deterministic data such as a sinusoidal wave, the latter methods with the model order of 3 or more can suppress bias errors if the data length is longer than one period.
- Variance errors of estimated peak heights are large for all of the methods. To suppress the errors sufficiently, these methods require sufficient number of samples similarly to the estimation of a general statistics.

The estimation test of the frequency-time spectrum was performed using simulated non-stationary data which change slowly peak frequencies of spectral component. The results indicate:

- The number of data samples (the length of time span) determines resolution of the time axis and also bias errors of the estimates. For the data which change the peak frequency or the peak height intermittently, the estimated spectrum seemingly decrease when the span length is long in consideration of the change of characteristics.

Finally, the demonstration test of this method was performed using real non-stationary data measured at Borssele reactor during the reactor shut-down operation. It was clearly shown that the stochastic properties of several signals change with decrease of coolant temperature and loop pressure. Furthermore, it was shown that the combined use of the frequency-time spectrum based on AR spectra and the time information functions, i.e. DC signal recordings and the instantaneous mean squares series makes it possible to analyze closely non-stationary data with respect to a time axis.

REFERENCES

Fig. 1 AR spectra estimated by four methods; $M=2$.

Fig. 2 AR spectra estimated by four methods; $M=3$.

Fig. 3 Bias errors for the peak frequency.

Fig. 4 Bias errors for the peak height.

Fig. 5 Block diagram of non-stationary signal generator.
Fig. 6 The simulated non-stationary data

Fig. 7 Power spectrum of the simulated non-stationary data; N=4096.

Fig. 8 Frequency-time spectrum of the simulated non-stationary data; N=1024.

Fig. 9 Frequency-time spectrum of the simulated non-stationary data; N=128.
Instantaneous AR spectrum

![Graphs of Primary Loop Pressure and Reactor Core Coolant Temperature](image)

Fig. 10(a) DC signal record; B11-243.  
Fig. 10(b) DC signal record; B11-244.

![Graphs of Ch.1 YA01.P002, Ch.3 XQ33.T01, Ch.5 YA01.P003, and Ch.7 YD01.V003](image)

Fig. 11(a) Instantaneous mean squares series; B11-243.  
Fig. 11(b) Instantaneous mean squares series; B11-244.
Fig. 12 Frequency-time spectrum of YA01.P002.

Fig. 13 Frequency-time spectrum of XQ33.T01.

Fig. 14 Frequency-time spectrum of YA01.P003.

Fig. 15 Frequency-time spectrum of YD01.V003.
SOME ASPECTS OF REACTOR NOISE ANALYSIS
FROM THE BAYESIAN VIEWPOINT

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ABSTRACT

This paper is concerned with a study of the fine structure of reactor noise. The power spectrum of reactor noise is represented in the frequency-time space using procedures of Bayesian AR-modelling (Kitagawa(1985), and Tamura(1985)). Results from the application to a number of power reactor noise data retrieve information on tomographical structures of peculiar reactor noise as well as apparently stationary reactor noise. Some comments are referred to the introduction of this idea to the STF analysis.

KEYWORD

Reactor noise; Anomaly; Nonstational Spectral Analysis; Frequency-time Power Spectrum; Bayesian Statistics; Akaike Bayesian Information Criterion(ABIC)

INTRODUCTION

Signal processing for reactor noise is performed basically over a wide length of data-span under a implicit assumption of stationarity. However, it happens sometimes to need detailed information on the local structure of a signal. This is particularly important for peculiar reactor noise signals which can not be regarded as stationary. Procedures of understanding this local behavior by a time evolution of power spectra defined on the segments of a signal have been already presented and used in many fields. Nevertheless, such an idea has been so far little applied in the reactor noise field.

Recently the methodology for nonstationary spectral analysis has been significantly developed. Concerning the nonparametric approach, the transformation of Wigner-Ville has enabled to obtain the instantaneous power spectrum(Martin, Flandrin(1985)), while concerning the time-series modelling approach, procedures of Bayesian modelling have been actively developed(Akaike(1979), Kitagawa(1983), (1985), Gersch(1985), Tamura(1984),(1985))

In this paper we present some examples of the fine structure of power spectrum in terms of a number of power reactor noise data, using two kinds of nonstationary univariate AR-modelling(Kitagawa(1985), Tamura(1985)). The power reactor noise data used are some of the data from the Borselle and the Phenix involved in the SNMTH III Benchmark Test.

BAYESIAN IDEA OF TIME SERIES ANALYSIS

The conventional(parametric) AR-modelling is concerned with a deterministic model whose parameters are determined uniquely. On the other hand, Bayesian AR-modelling is concerned with a stochastic model whose parameters are not determined uniquely. So the former considers only observation noise, but the latter both observation noise and system noise. Consequently, the policy of the ventional AR-modelling is to determine the variance of observation noise under minimization of AIC, while that of Bayesian modelling is to determine a trade-off of the variance of observation noise to the variance of system noise under minimization of a Bayesian version of AIC, i.e., ABIC(Tanabe, 1985). A method of Bayesian AR-modelling is basically stated as follows:

\[ y_k = \sum_{i=1}^{M} a_i y_{k-i} + \epsilon_k \]  

where \( a_i \) means a random parameter, \( y_k \) data, \( M \) model order, and \( \epsilon_k \) means white noise with mean 0 and
variance $\sigma^2$. Furthermore the parameter $a_i$ is imposed with a constraint on smoothness: $d^2a_i = \text{white noise}$, which means that the $d$-th order difference of a parameter behaves random walk.

The posterior distribution on parameter, $p(a/y, \delta, M, \sigma, \tau)$ is given by the prior distribution on parameter $g(a/d, \sigma, \tau)$ and the distribution on data $p(y/a, M, \sigma)$:

$$p(a/y, d, M, \sigma, \tau) = \frac{p(y/a, M, \sigma)g(a/d, \sigma, \tau)}{\int p(y/a, M, \sigma)g(a/d, \sigma, \tau)\,da}$$  \hspace{1cm} (2)

where $a = (a_1, \ldots, a_N)$, $y = (y_1, \ldots, y_N)$, and $\tau^2$ is the trade-off parameter above-mentioned. The likelihood of a Bayesian model is defined by

$$L(d, M, \sigma, \tau) = \int p(y/a, M, \sigma)g(a/d, \sigma, \tau)\,da$$

Then Akaike Bayesian Information Criterion (ABIC) is given by

$$\text{ABIC} = -2\log(\max_{\sigma} \log L(d, M, \sigma, \tau)) - 2(\text{number of parameters})$$  \hspace{1cm} (3)

Parameter estimation is performed as follows: 1) Maximize the numerator of Eq.(2) with respect to $a$, then estimates $\hat{a}$ and $\hat{\sigma}^2$ are obtained. 2) Minimize ABIC with respect to $d$, $M$, $\sigma$, then estimates $\hat{d}$, $\hat{M}$, $\hat{\sigma}$, are obtained, from which optimal parameter estimates are derived. Computations for optimization are made by the parameter search on $d$, $M$, $\sigma$. The power spectrum is obtained in the same way as with the conventional AR-modelling from the optimal parameter estimates except for the instantaneous power spectrum.

**BAYESIAN APPROACH FOR NONSTATIONARY SPECTRUM.**

We have briefly referred to a couple of procedures of nonstationary, more pertinently speaking, local stationary AR-modelling which we have used for studying the fine structure of reactor noise. The one was developed by Kitagawa(1985), Kitagawa and Gersh(1985), Kitagawa(1985). This approach is based on the time-varying AR-modelling, in which a different AR-model is fitted at each sampling time point. The model involved is given by putting $a_i$ as $a_{i,k}$:

$$y_k = a_{i,k}y_{k-1} + \epsilon_k$$  \hspace{1cm} (4)

$$\nu^d a_{i,k} = \delta_{i,k} \quad \text{Gaussian White Noise}$$  \hspace{1cm} (5)

In this procedure minimization of the ABIC(3) is not performed. Instead, the constraint model: for a time-varying parameter is represented as a state-vector equation. Application of the Kalman filter to the state-space representation enables to compute the likelihood on the data $y$ effectively. So maximization of the likelihood with $M$ and $d$ fixed determines parameters and trade-off parameters. the estimates of model order $M$ and difference order are estimated by minimization of the AIC. The instantaneous power spectrum of the time-varying parameters is defined by

$$P(r, k) = \frac{\sigma^2}{\sum_{i=1}^{M} a(i, k) \exp(-2j\nu)}$$  \hspace{1cm} (6)

The other was developed by Tamura(1984, 1985). This is an orthodox Bayesian approach, but considerably simplified compared with the one proposed by Akaike(1979). A given length of data-span is subdivided into $p$ blocks. The model order $M$ is assumed to be the same over each block. A similar assumption holds for the variance of residual. The AR-model over the $p$th block is given by

$$y_k = \sum_{i=1}^{M} a_{i,p}y_{k-i} + a_{p,0} + \epsilon_k \quad k = (p-1)r + 1, \ldots, pr.$$  \hspace{1cm} (7)

where $r =$ data-size of each block. The constraint on the smoothness for the temporally changing parameters is given by

$$\nu p_{,k} = \sum_{i=1}^{M} (a_{p+1,i} - a_{p,i})y_{k+i-m} + a_{p+1,0} - a_{p,0}$$  \hspace{1cm} (8)

$$E(v_{p,k}) = 0, E(v_{p,k}v_{p',k'}) = (\sigma^2/\nu^2)\delta_{p,p'}k,k'$$  \hspace{1cm} (9)

The above constraint means the difference between the error of one-step prediction with $(a_{p+1,i})$ and that with $(a_{p+1,i})$ over the $p$th block. In this procedure the data $y$ are included in the prior parameter distribution $g$, so that the likelihood is not regarded as marginal likelihood in the sense of Akaike(1984). The parameters $a_0$ over the first block are considered as trade-off parameters. Hence, maximization of the likelihood $L$ with respect to $a_0$ gives the estimate of the
model order M.

LOCAL STRUCTURE OF REACTOR NOISE VERIFIED BY BAYESIAN APPROACHES

In this section the local structures of reactor noise patterns from three kinds of categories are examined from the nonstationary AR-modellings above-introduced. Before going into the main subject, we define an instantaneous power spectrum and a local power spectrum. If given a data-span, a spectrum is defined at every sampling point belonging to the data-span, it is referred to the instantaneous power spectrum; if given a data-span, a spectrum is defined over a set of ordered, mutually disjoint segments of the data-span, it is referred to the local power spectrum. If stationarity is defined such that at any two adjacent points or adjacent segments each power spectrum is equal, then given a data-span, instantaneous stationarity is stronger than local stationarity.

The three kinds categories consist of 1) the local structure of apparently stationary signals, 2) the local structure of nonstationary signals with mean 0, and 3) the local structure of signals with a significant structur.

LOCAL STRUCTURE OF APPARENTLY STATIONARY SIGNALS

Figure 1A and Figure 2A indicate a pattern of Borsele excore neutron noise signals (log) and the corresponding instantaneous power spectrum computed by Kitagawa-Gersch's procedure based on a time-varying AR-modelling, respectively. Computations were made with the length of data-span=1000 and sampling time=0.02 s. The instantaneous power spectrum is given with the model order M=8 and the difference order d=2. The time evolution of instantaneous power spectrum reveals that given the data-span, the signal pattern reaches instantaneous stationarity after 16 s. The procedure used tackles effectively the structural change of a signal. Although the signal pattern concerned would not belong to a class of nonstationary signals, the time-evolving instantaneous spectra present useful information to the signal characteristics.

LOCAL STRUCTURE OF NONSTATIONARY SIGNALS WITH MEAN 0

Figure 2A and Figure 2B indicate a pattern of Borsele pressure noise signals and the corresponding instantaneous power spectrum obtained by Kitagawa-Gersch's procedure, respectively. From observation of Figure 2B, we can understand that time-evolution of the peak to a lower frequency region characterizes each wave packet of signal pattern. Although this type of signal should be analysed as a nonlinear time series, too, more detailed investigations by this nonstationary AR-modelling seem to bring interesting information to this signal pattern.

LOCAL STRUCTURE OF SIGNALS WITH SIGNIFICANT STRUCTURAL CHANGE

Among the signal patterns from the reactor Phenix, the secondary inlet temperature noise data indicates an interesting signal structural change. So we have examined its local structure, together with other noise data which have possibility to relate with it. The procedure used is the Bayesian nonstationary AR-modelling developed by Tamura. Although in this case a multi-variate-modeling was desirable, computations were based on the univariate modelling. The basic data for computing are

- Sampling Time
- Frequency Band
- Data Length/Block
- Total Number of Block
- Max. Setting Order for ARModel
- Total Length of Data-Span
- Statistical Signature

Power Spectrum

Figure 3A, Figure 3B and Figure 3C represent the Phenix secondary inlet temperature noise data (Ch. 10), the Phenix outlet temperature noise data of subassembly (Ch. 2) and the Phenix pump inlet temperature noise data (Ch. 6), respectively. The corresponding time-evolving power spectra are shown in Fig. 4A, Fig. 4B and Fig. 4C, respectively. These figures tell us that time-evolution of the peak of each spectrum before and after the structure change of signal present very nice information on the behavior of signal. Actually, in the case of the secondary inlet temperature noise the return to local stationarity is considerably delayed, whereas in the case of the pump inlet temperature noise this occurs immediately after the structure change of signal. Moreover, the temporal change of signal is synchronized in these signals. Similar results are observed on the Phenix Excore Neutron noise (Ch. 3) and the Phenix Flow noise, in Fig. 5A - Fig. 6A and Fig. 5B - Fig. 6B.
DISCUSSIONS

Some comments are referred to the two kinds of procedures for nonstationary spectral analysis, which we have applied to a number of reactor noise. The procedure of time-varying AR modelling (Kitagawa, Kitagawa-Gersch) divides by matching with envelope function representing the variance of data which is delivered as output. The point of partition may be selected with aid of a anomaly point arising in the envelope function. If the data-span is not divided in spite of the existence of anomaly points, the temporal change of the power spectrum decreases by the Kalman filtering. So in such a case the Borselle pressure noise presents a time-evolution of instantaneously stationary power spectrum. However, this partitioning problem seems to be not perfectly solved. On the other hand, in the Bayesian nonstationary AR-modelling (Tamura) trade-off parameter $\gamma$ is generally set in the region such as $10^{-5}$ to $10^4$. However, in noise signals whose variance of observation signal changes significantly, minimization of the $ABIC$ selects a large trade-off parameter so as to decrease the system noise. Therefore, the temporal change of the power spectrum does not likely appear. In order to identify such noise signals as the Borselle pressure noise or some of the Phenix noise data shown above, the value of $\gamma$ should be selected as smaller, for instance, $\gamma < 5$. However, this critical value has not been settled.

Extension of the univariate Bayesian AR-modelling to the multivariate modelling will increase the number of trade-off parameters. So parameter search will be complicated. However, its realization will bring more useful information to the stp analysis.

CONCLUSION

Identification of the local structure of a signal is very important, since this may be useful as a knowledge base for the anomaly detection. To do this end, we have performed to apply procedures of nonstationary spectral analysis recently developed by some authors. How to interpret noise signals still present difficult problems.

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The authors would like to appreciate greatly to Professor H. Akaike, Dr. G. Kitagawa and Dr. Y. Tamura for favorable suggestions.

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Fig. 1A Borssele Excore Neutron Noise (log)

Fig. 1B Borssele Pressure Noise (YAO1, P001)

Fig. 2A Time-evolving Instantaneous Spectrum of Borssele Excore Neutron Noise
- Computations were made by a procedure of G. Kitagawa (1985) -
Fig. 3A Phenix Secondary Inlet Temperature Noise (Ch. 10)

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- Computations were made by a Procedure of Y. Tamura (1985) -
Fig. 5A Phenix Excore Neutron Noise (Ch. 3)

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Fig. 6A Time-evolving Local Spectrum of Phenix Excore Neutron Noise

Fig. 6B Time-evolving Local Spectrum of Phenix Flow Noise of Primary Pump

- Computations were made by a procedure of Y. Tamura (1985) -
EXPERIENCES OF REACTOR NOISE DIAGNOSTICS
APPLYING PARAMETRIC SPECTRAL ANALYSIS
METHODS

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Abstract – The signal processing technology, like the choice of
parametrisation, methods of structure and parameter estimation,
computation of special frequency functions, that are available in
the reactor noise surveillance system of NPP Paks is reviewed.
A method for decomposition of the spectral densities to study
the effect of common source noises in multivariable situation
is also proposed. The paper presents experiences using the above
noise analysis methods using measurements on the 2nd and 3rd re-
actor units of NPP "Paks".

Keywords – reactor noise; noise source identification; ARMA
modelling

1. INTRODUCTION

The informations obtained from the measurements of process or noise signals of
nuclear power plants /NPP/ can be effectively used to detect failures or mal-
functions in the operation of the main components of the plant, like in the
operation of primary circuit or reactor intervals. In order to process these in-
formations, noise surveillance systems /NSS/ have been developed and applied in
many NPPs.

The surveillance of a system is usually based on the measurements of process sig-
nals, and on the conversion of the signals into a form in which the relevant sig-
nal properties can be analysed.

In earlier approaches the process or noise signals were converted into power spec-
tral densities and associated functions of frequency by applying e.g. the well
established technics of FFT.

More recently a considerable development can be observed in the use of time domain
and parametric spectral analysis methods based on AR and ARMA methods, which are
particularly powerful for the analysis of complicated multivariable systems.

The use of AR modelling pretends to be the most widely spread, perhaps because the
noise analysis can be supported by Noise Power Contribution /NPC/ and Signal Trans-
mission Path /STP/ analysis, see e.g. Oguma and Türkcan (1985). By the years of
eighteens, however, the ARMA modelling /structure and parameter estimation /
technics have become also well established and available, so it is expected that
their application can substitute the more limited AR modelling technics whenever
it is necessary. In addition, the generalization of NPC and STP analysis for the
ARMA case can also be solved, see e.g. the concept of Signal Effect Analysis /SEA /
in Veres et al.(1987), and the method of Spectral Decomposition /MSD/ discussed
later in this paper.

In the next section the signal processing methodology applied in the NSS of NPP
Paks will be shortly reviewed. This is followed by the characterization of the
setup of the NSS together with the description of experimental situations inves-
tigated in the paper. The results of AR, ARMA modelling and the properties of the
associated frequency functions are analysed in the last section.

2. SIGNAL PROCESSING METHODOLOGY APPLIED IN REACTOR NOISE ANALYSIS

The methods applied in reactor noise analysis will be reviewed. We focus our
attention to the application of parametric signal processing and spectral analy-
sis methods, thus classical nonparametric, like correlation and FFT-based spectral
analysis methods will not be considered here.

Reactor noise can be regarded as multivariable stochastic processes modelled as
the output of a linear filter driven by vector valued white noise sequence. Assume,
that $y_t$ is a $p$-dimensional discrete-time Gaussian process having rational Hermitian spectral density $\Phi(z)$, $z = \exp(i\omega)$. Assume that $\Phi(z)$ is positive definite for all $\omega \in [\pi, \pi]$, then applying spectral factorization, one can obtain a transfer matrix representation of $y_t$:

$$ y_t = T(z)e_t, \quad e_t \sim N(0,R), $$

(1)

where $e_t$ is Gaussian white noise sequence and $T(z)$ is a rational transfer matrix. Applying the results of stochastic realization theory, $T(z)$ can be realized in state-space, matrix fraction or ARMA forms. Since the use of transfer matrix and ARMA models are superior in the spectral analysis context, we consider these parametrizations only. ARMA models can be written in the form

$$ A(D)y_t = C(D)e_t $$

(2)

where $A(D)$, $C(D)$ are coprime nonsingular polynomial matrices with $A(0) = C(0) = I_p$, where $I_p$ is a $p \times p$ dimensional unit matrix and $D$ is the delay operator.

The polynomial matrices $A,C$ of the above monic ARMA models can be derived from the Hankel-matrices associated with the transfer matrix $T(z)$, see Gevers (1986) or from the constructibility matrix associated with $T(z)$, see Bokor and Keviczky (1987). The (McMillan) degree of an ARMA model is defined by $n = \deg \det A(D)$.

The irreducible ARMA models in Eq.2 are generally not uniquely identifiable except they are in canonic or pseudo canonic form, see Gevers and Werts (1984) for this important issue. The structure of canonic and pseudo canonic ARMA forms are uniquely defined by the so called structural indices, $n_i$, $i = 1, \ldots, p$, $\sum n_i = n$.

AR models are obtained from Eq. 2 if specifically $C(D) \equiv I_p$. The discussion of the structure and the computation of McMillan degree of AR models can also be found in Gevers (1986).

It should be noted, that in signal processing applications the use of fully parametrized AR models ($n_1 = n_2 = \ldots = n_p$) is the most common one.

In the remaining of this paragraph we shortly review the structure and parameter estimation methods associated with the above model classes available in the "method-base" of the noise analysis system. Since the computation of spectral functions from the identified models is fairly straightforward, these will be omitted here.

Transfer matrix models
The parameter estimation can be solved by the application of Maximum Likelihood (ML) method. It is also possible to apply various statistical tests to the residuals to validate the model.

ARMA models
In this noise analysis system the ML and Extended Least Squares (ELS) methods are applied for parameter estimation of ARMA models.

Using ML method, the structure estimation (determination of the structural indices from the observations) can be solved by applying the information criteria BIC or $\varnothing$, see e.g. Hamman and Kavalieris (1984).

Using ELS method, the procedure suggested by Bokor and Keviczky (1982) can be applied. This procedure fully exploits the structural properties of monic canonical or pseudo canonical ARMA forms and applies an F-test to the residuals.

AR models
A variety of Least Squares (LS)-type method has been elaborated for parameter estimation of AR models, or for estimation of AR parameters in ARMA models. Using fully parametrized AR models the parameter estimation can be solved recursively for model structures with increasing degree. This can be solved by applying fast lattice filters, see e.g. Whittle (1963), Haykin (1979), Friedlander (1982).

Since the lattice-filters are recursive in structure, it is easy to apply criteria like Final Prediction Error (FPE) or AIC for structure estimation. The methods available in the noise analysis system are based on the Whittle-algorithm, on the lattice algorithm for MEM, and a recently developed lattice algorithm for ARMA parameter estimation.
Frequency domain analysis of signals

The analysis of frequency domain characteristics of the signals is supported by the calculation of spectral densities and associated functions. Using the identified transfer function or AR, ARMA models, the spectral densities can be computed as

\[ S(\omega) = T(z) RT^*(z^{-1}), \]

where in the ARMA-case \[ T(z) = A^{-1}(z) C(z). \]

For solving special problems like noise source identification, causality analysis and detection of feedbacks, various frequency functions, like NPC, partial coherence, local effect, global effect functions, are also computed. The local- and global effect functions are utilised in the Signal Effect /SE/ analysis, see Veres et. al.,(1987). The NPC functions defined originally for AR model structures can also be generalised for transfer function and ARMA models as:

\[ GNPC_{ij}(\omega) = \frac{T_{ij} R_{jj} T_{ij}^*}{S_{ii}}, \]

where \( T_{ij}, T_{ij}^* \) denote the \( i,j \)-th elements of the transfer matrix \( T \) and its transposed-conjugate respectively, \( R_{jj} \) denotes the \( j \)-th element on the main diagonal of the source noise covariance matrix, \( S_{ii} \) denotes the autopower density APSD of \( i \)-th signal.

The physical meaning of \( GNPC_{ij} \) functions is the contribution of the inherent source noise of signal \( j \) to \( i \) \( i \) the spectral power of signal \( i \).

Studying the effect of common source noises in multivariable situations, it is useful to consider the Common Noise Power Contribution /CNPC/ functions defined by

\[ CNPC_{i\rightarrow j,k}(\omega) = \frac{T_{ij} R_{jk} T_{ik}^* + T_{ik} R_{kj} T_{ij}^*}{S_{ii}}. \]

The CNPC \( i\rightarrow j,k \) function can be interpreted as the contribution of common source noise of \( j \)-th and \( k \)-th signals to the spectral power of the \( i \)-th signal.

It is easy to deduce, that the following equality for the spectral density \( S_{ii}(\omega) \) is valid:

\[ S_{ii}(\omega) = \sum_{j} S_{ii}(\omega) \text{GNPC}_{ij}(\omega) + \sum_{j \neq k} S_{ii}(\omega) \text{GNPC}_{i\rightarrow j,k}(\omega), \]

which is called the spectral decomposition of \( S_{ii}(\omega) \) based on the GNPC and CNPC functions. As will be demonstrated in the next paragraphs, in the graphical analysis of the decomposed \( S_{ii}(\omega) \), the normalised version of Eq.6. is also very useful:

\[ \sum_{j} \text{GNPC}_{ij}(\omega) + \sum_{j \neq k} \text{GNPC}_{i\rightarrow j,k}(\omega) \equiv 1. \]

The analysis based on the decomposition of spectra using Eq.5-6 is called the method of Spectral Decomposition /SD/. In many practical situations, the changes in noise source behaviours cannot be well identified in the spectral densities themselves, since the spectral density of a given signal is influenced by each source noise. Thus in real multivariable situations the decomposed spectral densities are more sensitive tools for diagnostics than the spectral densities themselves.

DESCRIPTION OF THE NOISE MEASUREMENTS SYSTEM AND EXPERIMENTAL RESULTS

Four PWR-type reactor units operating with a nominal electric power of 440 MW have been installed in NPP Paks. The 1st reactor unit started its operation in 1980 and the 4th one started in 1987.

Equipments of noise diagnostic chains have been installed for each reactor units. The units 1st and 2nd have a common monitoring system, and so do units 3rd and 4th.
The scheme of hardware configurations of the diagnostic system for units 3rd and 4th is illustrated on Fig.1.
The noise measurements used to identify the models and frequency functions in this paper were performed in steady state operation on the 2nd and 3rd units operating on 100% reactor power.

The measured signals were selected into the groups given in Table 1, to investigate the dynamics of nuclear steam supply system, control rod driving mechanism and the effect of their dynamics on the neutron fluctuation.

The spectral functions associated with model A are shown on Fig.2. The spectral peaks appearing in the APSD-s of the signals S5, S7, S9 at frequencies around 3.5 Hz and 18.1 Hz can be associated with the first and second eigenfrequencies of the control rod vibrations. The broad spectral peak dominant in the frequency interval 8-12 Hz express the superposition of various effects. The shape of APSD-s in this frequency interval can be investigated by the application of spectral decompositions described by Eq. 5–7. See also the explanation of Fig. 3-4. later.

Table 1. Selection of signals for model identification and spectral analysis.

<table>
<thead>
<tr>
<th>2nd reactor unit</th>
<th>3rd reactor unit</th>
</tr>
</thead>
<tbody>
<tr>
<td>Model</td>
<td>Output Variables</td>
</tr>
<tr>
<td>A</td>
<td>S5</td>
</tr>
<tr>
<td></td>
<td>S7</td>
</tr>
<tr>
<td></td>
<td>S9</td>
</tr>
<tr>
<td></td>
<td>SP2</td>
</tr>
</tbody>
</table>

S1, i=5,7,9: acceleration of vibration on the control rod driving mechanism
R1, i=1,2,4: acceleration of vibration on the main reactor screws
T1: acceleration of vibration on the surge line between the pressurized and the primary circuit
SP2: in-core neutron flux fluctuation measured at coordinates 15-32 and reactor core height 2000 mm.

It can also be observed, that the spectral peaks in the APSD of neutron flux fluctuation SP2 appear at frequencies 6.6 Hz, 10.2 Hz, 14 Hz and 18.1 Hz respectively.

The analysis of NPC functions indicate that the noise sources at control rod vibrations have a strong effect on the neutron flux fluctuation at the frequencies 6.6 Hz, 10.2 Hz and 18.1 Hz. The effect in the opposite direction, i.e. the effect of source noise (resulted probably from the core barel or fuel assembly motion) associated with the neutron flux signal appears at frequencies 6.6 Hz and 18.1 Hz.

Studying the partial coherence and NPC functions of S5, S7, S9 signals, one can observe the existence of a strong common source noise. This observation is stressed also by the spectral decompositions of the APSD-s shown e.g. for the S5, S7 signals on Fig. 3. and Fig. 4, where three peaks can be observed in the frequency interval 8-12 Hz. The spectral peak at frequency 10.2 Hz was identified as the eigenfrequency associated with the pendulum-like motion of steam supply system.

The evaluation of the frequency functions defined by the spectral decomposition method will be illustrated using model B, where the interrelations among the signals S7, T1 and SP2 (see Table 1. for explanation) can be investigated. The decomposition of the APSD of signal S7 is shown on Fig.5 and Fig.6. shows the normalised decomposition. It can be observed, that in the frequency interval 8-12 Hz/where a broad peak appears in the APSD of S7/, approximately the 80% of this APSD is provided by the common source noises of the signals S7, T1 and S7, SP2 respectively.

The strongest effect of the common source noise of S7, T1 appears at the frequency 10.2 Hz (see Fig.5.), supporting the assumption that this common source noise might be resulted from the pendulum-like motion of reactor-primary circuit system.
Fig. 1. Scheme of hardware configuration of reactor noise diagnostic system on the 3rd and 4th reactor unit of NPP "Pakso".

Fig. 2. The spectral functions computed from model A.
Fig. 3. Spectral decomposition of S5 signal in model A.

Fig. 4. Spectral decomposition of S7 signal in model A.

Fig. 5. Original spectral decomposition of S7 signal in model B.

Fig. 6. Normalized spectral decomposition of S7 signal in model B.
Finally the spectral functions APSD-s GNPG-s and normalised spectral decompositions of APSD-s/computed from the identified multivariable ARMA (4,4) model of the acceleration of vibrations of main reactor screws model C/ are shown on Fig.7-9, for illustration. The spectral peaks, the cross- and feedback effects can be evaluated in a similar manner as it was characterized in case of model A and B.

Fig.7. APSD-s of the acceleration signals of the main reactor screw vibrations R1,R2,R3,R4 computed from the ARMA(4,4) model C.

Fig.8. Generalised noise power contribution functions computed from the ARMA(4,4) model C.
Fig. 9. Normalised decomposition of APSD-s computed from the ARMA (4,4) model C.
4. CONCLUSION

The paper presented experiences obtained from using parametric noise analysis methods available in the reactor surveillance system of NPP Paks. The methods included multivariable AR and ARMA modelling identified from sampled reactor noise signals/neutron and pressure fluctuations, acceleration of vibrations of control rods and reactor main crews. The models were used to compute various spectral function to investigate the frequency domain property of the signals. A method for decomposition of APSD-s was also applied to study the effect of common source noises in multivariable situations.

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SIGNAL EFFECT ANALYSIS FOR NOISE DIAGNOSTICS OF A NUCLEAR POWER PLANT

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Abstract - In this paper the feedback and STP (Signal Transmission Path Analysis) of linear systems is extended to a combined theory of signal analysis called SEA (signal effect analysis). Definitions are given for partial and global effects with their frequency decompositions. The analysis is illustrated both on simulative and real data measured in a nuclear power plant.

1. INTRODUCTION

Recently some methods have been proposed for multivariable source noise and feedback analysis. Oguma and Türcan (1984) presented a method called "Signal Transmission Path" (STP) analysis as a unified tool for noise source identification, signal causality analysis and detection of feedback. Another approach to reactor noise analysis is the "Dynamic Data System" (DDS) investigated by Wu and Ouyang. (1982) The definitions of linear dependence and feedback in terms of spectral analysis were given by Geweke (1984). Considering the above cited papers there is a close relation among these approaches. STP, DDS analysis and the investigation of linear dependence relations exhibit numerous similarities. Our first aim in the present paper is to show the relations among these approaches. The detailed discussion of the properties and definitions suggest some refinements of definitions. These refinements naturally provide some extensions of previous methods, leading to a new analysis procedure called Signal Effect Analysis (SEA). The efficiency of these methods will be illustrated by the analysis of reactor pressure and incore neutron flux fluctuation processes.

2. SOURCE NOISE AND FEEDBACK ANALYSIS

The main characteristics of the above cited approaches is the use of some parametric linear model of the system, such as multivariable AR or ARMA models. Assume a model

$$A(z^{-1})x_t = B(z^{-1})e_t$$

(2.1)

is fitted to the observed sequence $x(1), \ldots, x(N)$. Let $n_{(2.1)} A(z^{-1})$ and $B(z^{-1})$ are polynomials of the backward shift operator $z^{-1}$. $A(z^{-1}) = 1 + z^{-1} A_1 + \ldots + z^{-p} A_p$ and $B(z^{-1}) = 1 + z^{-1} B_1 + \ldots + z^{-q} B_q$. Let $\Sigma$ be the estimated covariance matrix of the "white noise" sequence $\{e_t\}$. It will be assumed here that the equation $\det (A(z)B(z)) = 0$ has all its roots outside the complex unit disc.

For the sake of simpler notations denote by $X_i(t)$ the Hilbert space spanned by $x_i(t-1), x_i(t-2), x_i(t-3), \ldots$. Use the notation $Pr(x/x_{i1}(t_1) \ldots x_{ik}(t_k))$ for the projection of a variable $x_i$ into the Hilbert space spanned by the variables $x_{i1}(t_1), \ldots, x_{ik}(t_k)$. Let $Var(x/x_{i1}(t_1) \ldots x_{ik}(t_k)) = D^2(x) - D^2(Pr(x/x_{i1}(t_1) \ldots x_{ik}(t_k)))$. Then the global effect between the components $x_i$ and $x_j$ is defined as

$$E(x_i \leftrightarrow x_j) = \ln \frac{Var(x_j(t)/x_j(t-1))}{Var(x_j(t)/x_i(t)), x_j(t-1))}$$

(2.2)

The instantaneous linear effect of the two variables is defined as the mutual information of the variables.
\[
x'_i(t) = x_i(t) - Pr(x_i(t)/x'_i(t-1), x_j(t-1)) \\
x'_j(t) = x_j(t) - Pr(x_j(t)/x'_i(t-1), x_j(t-1)),
\]

i.e.
\[
EI(x'_i, x'_j) = 2 \cdot I(x'_i, x'_j) = \ln \frac{\text{var } x'_i(t) \cdot \text{var } x'_j(t)}{\det \text{var } [x'_i(t), x'_j(t)]}
\]

The measure of linear dependence is then given by
\[
D(x'_i, x'_j) = E(x'_i \rightarrow x'_j) + EI(x'_i, x'_j) + E(x'_j \rightarrow x'_i)
\]

These definitions of linear dependence are slight modifications of those given by Geweke (1984). The only essential difference is that in the conditional part of (2.2) also the present of \( x_i \) is included. Furthermore, considering its relation with the modified noise power contribution function given in Definition 1, one may feel that our approach is more natural and practically useful. Instead of (2.2) Geweke (1982) has used the definition

\[
E(x'_i \rightarrow x'_j) = \ln \frac{\text{var } x'_j(t)/x'_i(t-1)}{\text{var } x'_j(t)/x'_i(t-1), x_j(t-1)}
\]

The transfer function of the model given in (2.1) is
\[
T(z^{-1}) = A(z^{-1})^{-1}B(z^{-1})
\]

Since \( \text{cov } (e_t) = \Sigma \), it is possible to consider another model with transfer function
\[
T'(z^{-1}) = T(z^{-1}) \Sigma_c^{-1}
\]

where the source noise covariance \( \Sigma_c \) may be the Cholesky factor of with \( \Sigma = \Sigma_c \Sigma_c^T \).

This gives the transfer function model
\[
x_t = T'(z^{-1}) e_t, \quad \text{cov } (e_t) = I
\]

If the observed process is two dimensional, \( x(t) = (x_i(t), x_j(t)) \), consider the combined infinite AR operator composed from the first row of \( T(z^{-1})^{-1} \) and from the second row of \( T'(z^{-1})^{-1} \), i.e. let
\[
G(z) = \begin{bmatrix}
T(z^{-1})^{-11} & T(z^{-1})^{-12} \\
T'(z^{-1})^{-11} & T'(z^{-1})^{-12}
\end{bmatrix}
\]

Then the effect of \( x_2 \) on \( x_1 \) at frequency \( \omega \) is defined by the components of the inverse of \( G(z) \)
\[
f_{x_2 \rightarrow x_1}(\omega) = -\ln \frac{|g_{11}(e^{j\omega})|^2}{|g_{11}(e^{j\omega})|^2 + |g_{12}(e^{j\omega})|^2}
\]

Geweke (1985) has proved that under fairly general conditions
\[
E(x_2 \rightarrow x_1) = \frac{1}{2\pi} \int_{-\pi}^{\pi} f_{x_2 \rightarrow x_1}(\omega) d\omega.
\]

Obuma (1982) defined the noise power contribution NPC function using \( T(z^{-1}) \) and for diagonal \( \Sigma \). To generalize Obuma's NPC function we introduce a new definition also for nondiagonal noise source covariance matrix.

Definition 1
\[
\text{NPC}(e_j \rightarrow x_i) = \frac{|T_{1j}(e^{j\omega})|^2}{\sum_k |T_{1k}(e^{j\omega})|^2}
\]
It is easy to see that (2.9) gives the same entity for diagonal $\Sigma$ as Oguma's (1984) definition given by

$$NPC_{ij} = \frac{|T_{ij}(e^{i\omega})|^2 \sigma_{jj}}{\sum_{k=1}^{r} |T_{ik}(e^{i\omega})|^2 \sigma_{kk}}, \quad \Sigma = \text{diag} (\sigma_{11}, \ldots, \sigma_{rr}).$$  \hspace{1cm} (2.10)

Consider a two dimensional process $x(t) = (x_1(t), x_2(t))$. Then another reasonable definition of the spectral effect of a variable on another variable is

**Definition 2**

$$h_{x_1 \rightarrow x_2}(\omega) = \ln \frac{|T_{22}(e^{i\omega})|^2 + |T_{21}(e^{i\omega})|^2}{|T_{22}(e^{i\omega})|^2}$$  \hspace{1cm} (2.11)

Now, we give short heuristic explanation of the above definitions.

Let:

1. $\sigma_{22}^2$ be the squared prediction error of $x_2(t)$ from its past observations $x_2(t-1), x_2(t-2), x_2(t-3), \ldots$.
2. Let $\sigma_{21}^2$ be the squared prediction error of $x_2(t)$ from its past and from the present and past of $x_1(t)$.

Then the decrease of the prediction error of $x_2(t)$ due to the knowledge of $x_1(t), x_1(t-2), \ldots$ can be considered as the formal linear effect of $x_1$ on $x_2$, and it can be measured by $\ln(\sigma_{22}^2 / \sigma_{21}^2) = \mathbb{E}(x_1 \rightarrow x_2)$.

The relation to the NPC function is characterized by

$$h_{x_1 \rightarrow x_2}(\omega) = \ln(1 - \text{NPC}(e^{i\omega} \rightarrow x_2))$$  \hspace{1cm} (2.12)

Observe that from the triangularity of the leading coefficient matrix on the right hand side of (2.6) it follows that the knowledge of $x_1(t), x_1(t-1), \ldots, x_2(t-1), x_2(t-2), \ldots$ is equivalent to the knowledge of $x_1(t), e_1(t-1), \ldots, e_2(t-1), e_2(t-2), \ldots$.

The following notes are parallel to (1') and give the explanations how to understand the effect of a variable on another variable at a given frequency $\omega$.

1' $\ln \sigma_{x_2}(e^{i\omega}) = \ln(|\hat{g}_{21}(e^{i\omega})|^2 + |\hat{g}_{22}(e^{i\omega})|^2)$ is a measure of indefiniteness of the variable $x_2(t)$ at frequency $\omega$. It is shown that the process $x_2(t)$ can be arbitrarily closely approximated in the mean squared sense by

$$x_2(t) = \sum_{j=0}^{\infty} \sum_{j=0}^{\infty} (\cos \omega_j t + i \sin \omega_j t)$$

where $\omega_j, j = 0, 1, 2, \ldots$ are such that $e^{i\omega_j t}$, $j = 0, 1, \ldots$, is a sequence of orthogonal functions. Then the spectral distribution of $x_2$ is approximated by the monotonically increasing step function with steps $E_{2j}$ at $\omega_j \in [-\pi, \pi], j = 0, 1, 2, \ldots$. In this way it can heuristically be seen that the spectral density characterizes the variance ("spectral power", "random intensity") of the spectral component of frequency $\omega_j$.

2' Continuing the above ideas it is fairly clear that $\ln |\hat{g}_{22}(e^{i\omega})|^2$ characterizes the spectral power of the variable obtained as $x_2(t)$ minus the best linear predictor of $x_2(t)$ based on the present and past of $x_1(t)$.

Thus, the difference

$$\ln |\hat{g}_{22}(e^{i\omega})|^2 - \ln |\hat{g}_{22}(e^{i\omega})|^2$$

gives the decrease in spectral power at frequency $\omega$, which is due to the knowledge of the present and past of $x_1$. \hspace{1cm} (2.13)
The parallel argumentations in (1), (2) and (1'), (2') can be connected by the following theorem.

**THEOREM 1.**

\[ B(x_1 \rightarrow x_2) = \int_{\mathbb{R}} h(x_1 \rightarrow x_2)(\omega) \, d\omega \]  

(2.13)

**Proof.** According to the above derivations it is sufficient to prove that

\[ \ln(\sigma^2_{21}/\sigma^2_2) = \int_{\mathbb{R}} \left[ \ln(|g_{21}(e^{i\omega})|^2 + |g_{22}(e^{i\omega})|^2)/|g_{22}(e^{i\omega})|^2 \right] d\omega \]

which is the consequence of the simple equalities

\[ \ln \sigma^2_2 = \int_{\mathbb{R}} \ln \mathcal{F}_{x_1}(e^{i\omega}) \, d\omega \]

and

\[ \ln \sigma^2_{21} = \int_{\mathbb{R}} \ln |g_{22}(e^{i\omega})|^2 \, d\omega \]

In the following we give the reasonable definitions of the partial linear effect and its spectral decomposition. The partial linear effect (PLE) and the partial noise power contribution (PNPC) functions can be defined by neglecting the influence of the remaining variables. This idea, which originates from the definition of partial correlation and partial coherence, is useful to measure the effects between the signals and between the noise source and signals, respectively.

The **global partial linear effect** (GPLE) is defined as

**Definition 3.**

\[ E(x_i \rightarrow x_j / x_k, k \neq i, j) = \frac{\text{var}(x_j(t) / x_i(t), x_j(t-1), x_k(t), k \neq j, i)}{\text{var}(x_j(t) / x_i(t-1), x_k(t), k \neq j)} \]  

(2.14)

Let

\[ G(z) = \begin{bmatrix} \left[T'(z^{-1})^{-1}\right]_{1i} & \left[T'(z^{-1})^{-1}\right]_{ij} \\ \left[T'(z^{-1})^{-1}\right]_{ji} & \left[T'(z^{-1})^{-1}\right]_{jj} \end{bmatrix}^{-1} \]

and define the spectral PLE function by

\[ h(x_i \rightarrow x_j / x_k, k \neq i, j)(\omega) = \ln \left| \frac{G(z)_{jj}^{-1}}{G(z)_{ij}^{-2}} \right| \bigg|_{z = e^{i\omega}} \]

(2.15)

Similarly to (2.12) for \( \Sigma = \text{diag}(\sigma_{ii}) \) there is a relation

\[ h(x_i \rightarrow x_j / x_k, k \neq i, j)(\omega) = -\ln(1 - \text{PNPC}(e_i \rightarrow x_j)(\omega)) \]

with the PNPC function defined by Oguma (1984) as

\[ \text{PNPC}(e_i \rightarrow x_j)(\omega) = \left| G(z)_{ij} \right|^2 \left| G(z)_{ij}^{-2} + G(z)_{ii}^{-1} \right| \bigg|_{z = e^{i\omega}} \]

(2.16)

Finally, for computing the global PLE the integral representation

\[ E(x_i \rightarrow x_j / x_k, k \neq i, j) = \frac{1}{2\pi} \int_{\mathbb{R}} h(x_i \rightarrow x_j / x_k, k \neq i, j)(\omega) \, d\omega \]

can be used.
APPLICATIONS

In this section we give some illustrative applications of Signal Effect Analysis for data measured by the Noise Diagnostic System at NPP "Paks". The measurements were performed under normal steady state operation in full reactor power. The signals to be analysed were chosen as follows:

In the 2nd reactor unit (model A)
1. S7 - vibration of control rod driving mechanism
2. T1 - vibration of pipe between pressurizer and primary loop.
3. SP2 - in-core neutron flux fluctuation

In the 3rd reactor unit (model B)
1. 3G5 - vibration of steam generator house N°5.
2. 3M5 - vibration of main valve in the primary loop N°5 (hot Cag)
3. 3P5 - vibration of the pump in primary loop N°5.
4. 3P5 - pressure fluctuation in the primary loop N°5.

In Fig.1 and Fig.2 the signals measured by the Noise Diagnostic Surveillance System are shown.

![Fig.1](image1)

![Fig.2](image2)

The measurements were performed with analog filtering (HP filters: 0.1 Hz and LP filters 20.0 Hz). The sampling frequency was 64 Hz.

The log-gains of the signals are shown in Fig.3. In Table 1 we give the most important spectral peaks to be observed.

![Fig.3](image3)

<table>
<thead>
<tr>
<th></th>
<th>spectral peaks (sp.p)</th>
<th>2S7</th>
<th>2T1</th>
<th>SP2</th>
<th>2S7</th>
<th>2T1</th>
<th>SP2</th>
</tr>
</thead>
<tbody>
<tr>
<td>2S7</td>
<td></td>
<td>4.6</td>
<td>10.8</td>
<td>17.8</td>
<td>-</td>
<td>3.44</td>
<td>9.13</td>
</tr>
<tr>
<td>2T1</td>
<td></td>
<td>5.4</td>
<td>9.4</td>
<td>17.8</td>
<td>3.03</td>
<td>9.6</td>
<td></td>
</tr>
<tr>
<td>SP2</td>
<td></td>
<td>6.6</td>
<td>14.2</td>
<td>18.3</td>
<td>2.75</td>
<td>5.8</td>
<td></td>
</tr>
</tbody>
</table>

Table 1. Spectral peaks of log-gains and global and partial effect.
In Table 1 the integrals of the spectral decompositions of signal effects and partial signal effects in the interval 0–20.0 Hz are summarized, which are called the global effects on interval /0.1–20.0 Hz/

It can be seen from Table 1 that the global effects between the vibration signals are approximately balanced, while the global effects of the two vibration signals on the in-core neutron flux fluctuation is 2–3 times larger than the reverse effect.

The values of the global partial effect show another picture. There seems to be a strong connection between in-core neutron flux fluctuation and control rod driving mechanism. The effect of the last one is dominant. The same is shown by global effect as it was to be expected on the basis of physical arguments.

The results of Table 1 are illustrated in Fig. 4.

Fig. 4 Graphs of global effects in frequency range 0.1–20.0 Hz

Fig.5 Frequency decomposition of the partial effect

A thorough analysis of the effects shows that a given signal shows a strong connection with other signals only at certain frequencies. This statement does not hold for non-partial effects. E.g. signal S7 shows a strong effect on both T1 and SP2 at frequency 3.8 Hz. This effect is significantly more on in-core neutron flux fluctuation. (Note that there are still some less significant effects at frequencies 8 Hz, 14 Hz and 20 Hz).

The effect of SP2 on S7 and T1 is strong at frequency 17.1 Hz. Furthermore, SP2 still has effect on S7 at about the frequencies 12 Hz, 8 Hz and 3 Hz, but these effects are not characteristic.

Signal T1 (vibration of pipeline between pressuriser and primary loop) shows a bit different behaviour. It effects neutron flux fluctuations practically only at low frequencies, and control rod driving mechanisms only at frequency 10.8 Hz, and it has an even stronger effect in the interval /0.5 Hz/. and at about 18.6 Hz. Observe that frequency 10.8 is already known from other measurements and it can be identified with the swinging movement of the whole primary loop.

The analysis of non-partial signal effect still shows a more varied picture. The characteristic frequencies are summarized in Fig.6.

Fig.6. Summarizing graph of the signal effects.
The spectral decomposition of spectral effects is shown in Figure 7.

Fig. 7: Spectral decompositions of signal effects between T1, S7, SP2.
In Fig. 6 the most significant peaks of effects are in parenthesis.
In a similar way we shortly give the analysis of model B.
The log-gain and PLE functions are given in Figure 8.

Fig. 8: Log-gain and partial linear effect functions.
The characteristic peaks of log-gains (and of ADSD functions) are summarised in Table 2.

<table>
<thead>
<tr>
<th>G5</th>
<th>7.6</th>
<th>12.1</th>
<th>16.7</th>
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<tbody>
<tr>
<td>M5</td>
<td>5.7</td>
<td>7.9</td>
<td>10.4</td>
</tr>
<tr>
<td>P5</td>
<td>6.4</td>
<td>7.6</td>
<td>12.1</td>
</tr>
<tr>
<td>P5</td>
<td>6.4</td>
<td>7.9</td>
<td>12.1</td>
</tr>
</tbody>
</table>

Table 2. Peaks in log-gains

The partial effect analysis gives the following results.

305 (vibration of the steam generator house does not affect neither the vibration of (3M5) (the main valve N°5) nor pressure fluctuation, however the latter has an effect in the reverse direction. At the same time at low frequencies (at about 6–7 Hz, 12.1 Hz and 17.5 Hz there is some effect on the vibration of main circulating pump. It is interesting to note that the latter effect is not at frequency 16.7 Hz.

3M5 (the hot leg main valve) affects the other three vibration signals in the environment of 12.1 Hz and 18 Hz. The amount of effect is in all the other three cases almost the same. It is interesting, since there are no peaks in the log-gain of the main valve at the expected frequencies. The main valve is a passive body concerning the vibration of the whole system, the movement of which is mainly determined by the constraints of its environment and it has a poor influence on its environment.

3P5 (vibration signal of the pump in primary loop N°5) mainly effects on the other three signals at frequency 3.2 Hz.

3P5 (pressure fluctuation in the primary loop N°5) has strong effects on the other three signals at frequencies 4.1 Hz, 12.1 Hz and 16.7 Hz. Frequencies 12.1 and 16.7 Hz seem to be eigen-frequencies of loop pressure fluctuation.

At these frequencies there are peaks also in the auto-spectra and at frequency 4.1 Hz there is no peak in the log-gain of 3P5.

The global values of effects are shown in Table 3.

<table>
<thead>
<tr>
<th></th>
<th>305</th>
<th>3M5</th>
<th>3P5</th>
<th>3P5</th>
</tr>
</thead>
<tbody>
<tr>
<td>305</td>
<td>-</td>
<td>0.0052</td>
<td>0.09677</td>
<td>0.0066</td>
</tr>
<tr>
<td>3M5</td>
<td>0.7181</td>
<td>-</td>
<td>0.6456</td>
<td>1.1032</td>
</tr>
<tr>
<td>3P5</td>
<td>0.6219</td>
<td>3.3064</td>
<td>-</td>
<td>1.5852</td>
</tr>
<tr>
<td>3P5</td>
<td>1.3985</td>
<td>0.5898</td>
<td>1.9455</td>
<td>-</td>
</tr>
</tbody>
</table>

Observe that the largest global effect on the vibration of the system is given by the main pump and pressure fluctuation in the given frequency interval.

CONCLUSION

This paper is a contribution to existing methods for the analysis of NPP processes. The presented SEA methodology gives generalization of both Oguma's NPC analysis and of Geweke's conditional feedback analysis. These methods were successfully applied to the investigation of relations among measured reactor signals.

REFERENCES


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SAFETY RELATED APPLICATIONS (PART II)

Session chairman: J. Valko (Hungary)
SUMMARY OF SESSION

Three papers were presented in this session, they all deal with BWR stability. Several methods of stability monitoring have been examined by the authors and the respective merits and drawbacks have been analysed. Two papers out of the three report on microcomputer based stability monitoring systems applicable in BWR control rooms.

Kanemoto et al. presented a paper which firstly compared several methods for estimating the reactor stability. Process noise signals such as neutron flux or core flow rate are used. Univariate and Multivariate Auto-Regression methods are used and the Decay Ratio (DR) is calculated by different techniques. The methods have been checked with simulated data and also with real power plant data. A BWR stability monitoring system was constructed and its capabilities were tested in a power plant.

In the paper presented by van der Hagen et al., the natural circulation cooled Dodewaard BWR was examined in 6 different experimental conditions. The methods and criteria used for assessing the stability of the BWR core have been compared in the 6 cases, perturbation was introduced through control rod movement or steam valve operation. The method of peaking factors (RMS value ratios in given frequency ranges) is suggested as a reliable yet simple technique.

Federico et al. presented their paper starting with a comparison of different methods and algorithms. The method adopted to the stability monitoring system is based on Univariate Auto-Regression, and the Decay-Ratio (DR) is evaluated. Artificial data were first used to analyse the capabilities of the method, then experimental data from the Caorso nuclear power plant were used.

In conclusion, the session showed that BWR core (channel) stability monitoring is an important safety related topic. By continuous monitoring, reactor operators can operate their reactor closer to stability limits and reduced margins can be allowed. This helps in load following and in other cases when the reactor is not running at nominal conditions. The decay ratio in the basic quantity to characterize BWR stability.

Microprocessor based on-line systems monitoring global and local stability and providing easy to read displays to operators have now been introduced and used successfully at several reactors.
DEVELOPMENT OF AN ON-LINE REACTOR STABILITY MONITORING SYSTEM IN A BOILING WATER REACTOR

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Abstract - Reactor stability is an important topic in BWR plant design and operation. The objective of the present study is to establish a method for estimating the reactor stability in an operating BWR by measuring process noise signals, such as neutron flux or core flow rate. In the present study, several methods, based on autoregressive modeling, are presented and their reliability is quantitatively evaluated, using both simulated and real plant noise data. On the basis of these examinations, an on-line stability monitoring system was developed. Through field tests on the system over a year in an operating BWR, the validity of the system has been confirmed. Also, valuable data about reactor stability has been obtained.

1. INTRODUCTION

In a boiling water reactor (BWR), reactor stability induced by thermal-hydraulic and void reactivity feedback has been one of important topics in plant design and operation. Extensive study has been made on the subject for many years, from analytical and experimental viewpoints. In these studies, the reactor stability is classified into two typical categories, (a) channel thermal-hydraulic instability, which is a local phenomenon in a fuel bundle, and (b) reactivity feedback instability, which is a global phenomenon in the reactor core. Furthermore, a regional reactivity instability phenomenon, which involves out of phase neutron flux oscillations in the reactor core was, reported in a recent paper (Gialdi, 1984). Precise understanding of these phenomena is very important.

The objective of the present study is to establish a method for estimating reactor stability in an operating BWR by measuring process noise signals, such as neutron flux or core flow rate. Experimental estimation of the reactor stability was mainly carried out with the help of an artificial perturbation test (Carmichael, 1978; Mitsutake, 1983). However, such a test cannot be easily accomplished in a commercial power plant. Hence, it is desirable to develop an easy and accurate stability estimation method. A noise analysis method is quite promising for this purpose. A number of methods to estimate the reactor stability from noise data has been reported (Kanemoto, 1984; March-Leuba, 1984; Wu, 1981; Upadhyaya, 1981). Most of these method are based on the autoregressive(AR) modeling method. However, their concrete methods differ from each other in regard to AR parameter estimation and the stability index estimation methods. In other words, the accuracy and reliability of these methods has not been fully established. In the present study, a number of these methods are mutually compared and their merits and demerits are clarified. Especially, by means of various simulation and real plant data analysis, the accuracy of the estimated stability index (the decay ratio) is quantitatively evaluated.

On the basis of the above consideration, a micro-computer based on-line stability monitoring system has been developed. The system can evaluate the stability margin in a real time. The results obtained in a field test on the system in an operating plant will be discussed.
2. STABILITY ESTIMATION METHOD

A number of stability estimation methods have been proposed in earlier studies using neutron noise signals. As one of these methods, the authors proposed a method based on the multivariate AR model (Kanemoto, 1984). The feasibility of the method was confirmed by analyzing both steady-state noise data and perturbation test data in an operating BWR plant. However, in order to practically use this method, it is desirable to develop an easier and more robust method. In this section, several methods will be presented for stability estimation.

2.1. Autoregressive modeling

A univariate AR(UAR) model of order M for the time series \( x(t) \) is of the form

\[
   x(t) = \sum_{k=1}^{M} a(k)x(t-k\Delta t) + v(t)
\]

where \( a(k) \) are AR coefficients and \( v(t) \) is a residual noise sequence. The AR coefficients are determined to minimize the variance of the residual sequence.

In a concrete parameter estimation from given time series data, several algorithms are possible. Here, three typical algorithms are considered, the Yule-Walker(Y-W) method, the Burg method, and the direct least squares estimation using Householder transformation. The Y-W method is based on the correlation function of the time series data and minimizes the forward prediction error for the AR model. Here, it should be noted that the correlation function calculated from finite data is an approximation especially for short length data. The Burg method is an improved Y-W method and minimizes both forward and backward prediction errors. The Householder method is the most exact one in the meaning that the AR parameters are directly calculated from time series data without using the correlation function. The difference in these three algorithms stands out when short length data are analyzed (Kitamura, 1984). Concrete comparison will be shown in the following section.

Determining model order \( M \) is also an essential problem in the stability estimation. The most accurate definition of stability may be the poles and zeros of the system equation, if the linear model can be assumed. However, when the univariate AR model is applied to this system, each zero in the real system is approximated by multiple ring poles arranged in a complex plane (Kishida, 1984). Hence, the characteristic roots of Eq.(1) do not have one to one correspondence to the poles in the real system. This suggests that care should be taken to determine the stability index from the characteristic roots in the univariate AR model. The circumstances are the same if the autoregressive and moving averaged model, which has multiple zeros and poles, is applied, since the true order of the system cannot be known. In the present study, the Akaike's information criterion (AIC) is applied to determine the model order (Akaike, 1974). This criterion was very effective in the authors experience, however, it should be noted that this criterion is just an approximation and does not correspond to the real system order.

2.2. Stability index estimation algorithm

The stability index is usually expressed by the decay ratio (DR), which is defined by the ratio of successive overshoot peak values. In order to estimate the decay ratio from Eq.(1), the following algorithms are possible.

2.2.1. Method based on impulse response function. The impulse response function can be calculated from Eq.(1) by

\[
   h(i) = \sum_{k=1}^{M} a(k)h(i-k)
\]

with initial conditions \( h(0)=1 \) and \( h(-i)=0 \). From the ratio of successive peak values of \( h(i) \), the decay ratio can be evaluated. Here, it should be noted that the peak ratio changes, depending on the selected peak positions, except for the second order system. Hence, two kinds of decay ratios can be defined, namely, the apparent DR and the asymptotic DR (March-Leuba, 1984). The former is defined by the ratio of the first two peak values, and the latter by the ratio of asymptotic peak values. In the authors' experience, asymptotic DR corresponds to DR calculated by the most unstable characteristic root of the UAR model. However, when lower frequency fluctuating components, mainly due to the control system characteristics, are mixed in neutron noise, this method often fails to properly estimate the DR, since the time domain approach is inadequate to remove fluctuating
components in a certain frequency region.

2.2.2 Method based on characteristic root. The decay ratio can be calculated from the most unstable characteristic root of the following equation:

\[ 1 - \sum_{k=1}^{N} a(k)z^k = 0. \]  

According to the authors' experience, this method is the most accurate, if model order \( N \) is properly determined. On the other hand, it should be noted that this method is very sensitive to the model order.

2.2.3 Method based on power spectral density (PSD). PSD can be calculated from Eq.(1) by

\[ P(f) = \frac{< v^2 >}{1 - \sum a(k) e^{-2\pi f \omega k}}. \]  

The sharpness of the resonance peak for this PSD, around 0.5 Hz, corresponds to the stability index. The simplest method to evaluate the DR from this peak is the so-called half-power bandwidth relation, namely,

\[ \Xi = \frac{\Delta f}{2 \pi f_r}, \quad \text{DR} = \exp \left( -2\pi \Xi \sqrt{1-\Xi^2} \right), \]  

where \( \Xi \) is the damping ratio, DR the decay ratio, \( f_r \) the peak frequency, and \( \Delta f \) the half-power bandwidth. This relation can easily be extended into the following relation. Assume that Eq.(3) can be approximated by

\[ P(f) = \frac{1}{(f^2 - f_r^2)^2 + f_0^4 - f_r^4}, \quad Q(F) = \frac{\Delta F}{2}. \]  

where \( f_r = f_0 \sqrt{1 - 2\Xi^2} \), and \( F = f^2 \). Next, assume the parameter by

(a) \[ \Xi^2 = \frac{1}{(1-2\Xi^2)^2} - 1 = \frac{2}{F^2}, \quad Q^* = \frac{\Delta F^2}{F^2} \]  

(b) \[ \Xi = \frac{1}{(k-1) f_r^4} \left( \frac{\Delta F}{2} \right), \]  

then, the damping ratio can be obtained by

\[ \Xi = \sqrt{\frac{1}{2} \left( 1 - \frac{1}{1+\Xi^2} \right)}. \]  

Here, \( \Delta F \) is the 1/\( k \) - power bandwidth. Since the right-hand sides of Eqs.(7a) and (7b) can be calculated from the observed PSD of Eq.(4), the DR can be calculated from Eq.(8). If \( k=2 \) and \( \Xi < 0 \) are assumed, Eq.(8) can be reduced into Eq.(5). Hence, the relation of Eq.(8) is called by an extended bandwidth (EBW) relation in the present paper. In the above relations, (a) suggests that the DR can be calculated from local information around the resonance peak of the PSD. This suggests that relation (a) corresponds to the above mentioned most unstable characteristic root. On the other hand, the DR calculated from (b) corresponds to an averaged value of the DR in a certain frequency region.

2.2.4 Method based on least squares fitting. The decay ratio can also be obtained by fitting Eq.(6) to observed PSD. In this case, if Eq.(6) is directly fitted to Eq.(4), a nonlinear fitting algorithm is needed. However, if \( Q(F) \) in Eq.(6) is fitted to the denominator of Eq.(4), a simple linear fitting can be used. The most important point in this method is the selection of a fitting frequency region. The objective criterion for this selection will be necessary.

Fig.1 Illustrative block diagram for BWR core dynamics.
2.3. Multivariate autoregressive modeling

Multivariate autoregressive (MAR) modeling is another important method in stability estimation. In the artificial perturbation test for stability estimation, the transfer function (TF) from the reactivity perturbation to neutron flux is used to evaluate the DR. Usually, the reactor pressure is used for this reactivity perturbation. In the previous work, one of the authors applied the MAR model to estimate the above TF and clarified that it was possible to obtain two kinds of TFs, namely, open-loop and closed-loop TFs (Kanemoto, 1984; Mitsutake, 1983). In the block diagram for the reactor dynamics, shown in Fig. 1, the open-loop TF includes the local void feedback (loop 1) in a fuel channel. On the other hand, the closed-loop TF includes the feedback (loop 2) through reactor pressure and core flow rate in the reactor pressure vessel. From a stability monitoring viewpoint in a real plant, the closed-loop TF is regarded as a more practical index, since it includes overall plant feedback loops and represents a real situation of the reactor stability. It should also be noted that the closed-loop stability index corresponds to that from the above UAR modeling.

Compared with the above perturbation test, the problem in the noise analysis method is how to evaluate input noise source characteristics. The MAR model also meets this need, since it can identify the input noise source and its transfer characteristics in a separate form. In the above work, the MAR modeling was applied to both steady-state noise data and artificial perturbation test data. In the results obtained there, both stability indices from noise and perturbation data agree well with each other for the open-loop and closed-loop TFs.

3. Examination of Stability Estimation Method

In the practical application of the above various stability estimation methods to operating plant data, it is necessary to evaluate merits and demerits for each method quantitatively. The practical method should meet pertinent needs, such as the maximum allowable error (bias and scattering) for the DR, minimum data length needed in the DR estimation, a robust algorithm to various data condition changes, etc. Here, the accuracy of DR estimation and its dependence on various estimation conditions will be discussed, using simulated noise data and real BWR plant noise data.

- Fig. 2 Block diagram for simulated noise generation.
- Fig. 3 Comparison of two algorithms for decay ratio estimation.

3.1. Simulation analysis

In this section, the above mentioned several stability estimation methods are mutually compared and their accuracy discussed, using computer generated random noise data. As shown in Fig. 2, the white noise is fed into the second order system and the shaping filter. Then, the output signal is analyzed by the above methods, based on the UAR model. Here, the shaping filter is added to simulate actual noise characteristics in a real plant. In the following discussion, the difference in concrete methods for the UAR modeling and DR estimation techniques will be examined.

At first, typical DR estimation algorithms are compared. In this examination, several kinds of shaping filter, shown in Fig. 2, are assumed. In Fig. 3, DR estimation results from ten kinds of simulation cases are compared. Here, two
kinds of DR estimation algorithms, the most unstable characteristic root and the EBW method with the parameter k=2, are applied to these data. The use of the characteristic root is considered to be most accurate, if the AR model order is properly determined. However, this method may be sensitive to the model order. On the other hand, the EBW method may be more robust to the model order, since this method is based on the global shape of the resonance peak in the PSD. The results, shown in Fig.3, support this presumption. In cases 3 and 4, the DRs estimated by the characteristic root are biased. The PSDs in cases 3 and 6 are shown in Fig.4 with the AR model order. In case 3, the model order is higher than in other cases. This fact causes bias on the DR, estimated by the characteristic root. On the other hand, the DRs, determined by the EBW method, are well evaluated for all cases.

In the next examination, dependences on the UAR modeling methods are discussed. In Table 1, two typical methods of the UAR modeling, Y-W and Burg algorithms, are compared. The dependence on the AR model order is also shown. Here, the DRs are estimated from 640 point data with 0.2 second sampling and averaged over 32 independent samples. In this simulation, the true DR is set to be 0.5 and the second order model without the shaping filter is used. In Table 1, the DR estimated by the Y-W algorithm is slightly low, when the AIC is used to determine the AR model order. This conclusion agrees with the previous work (March-Leuba, 1984). They concluded that the DR estimated by the UAR model was biased when the data length was short. However, in the case of the Burg algorithm, this conclusion does not hold. Furthermore, even in the Y-W algorithm, the DR can be estimated, if the model order is properly selected. The above simulation analysis suggests that the determination of the model order is very important, especially, when short length data are used.

<table>
<thead>
<tr>
<th>Model Order</th>
<th>Y-W</th>
<th>Burg</th>
</tr>
</thead>
<tbody>
<tr>
<td>A I C</td>
<td>0.480±0.060</td>
<td>0.560±0.086</td>
</tr>
<tr>
<td>M = 2</td>
<td>0.395±0.039</td>
<td>0.502±0.056</td>
</tr>
<tr>
<td>M = 10</td>
<td>0.593±0.071</td>
<td>0.518±0.073</td>
</tr>
</tbody>
</table>

Table 1: Dependence of decay ratios on the AR modeling algorithm.

Fig.5 Dependence of decay ratio estimation accuracy on data length.

Fig.6 Dependence of decay ratios on the AR modeling parameters.

Finally, the estimated DR accuracy dependence on the data length is examined. Figure 5 indicates the results obtained by two kinds of simulation. The DRs in Fig. 5(a) are obtained from 128 second data with 0.2 second sampling and averaged
over 32 samples. The DRs in Fig.5(b) are obtained from 512 second data with and averaged over 8 samples. In both data sets, the true DRs in the simulation model are assumed to be from 0.2 to 0.9. It can be seen, from these figures, that the average values of the estimated DR agree well with the true ones in both cases. On the other hand, the scattering in the estimated DR depends on the data length. It should also be noted that the scattering does not depend on the true DR value, itself. In a practical use of the stability estimation methods, a trade-off between the maximum allowable error (scattering) and minimum data length will be important.

3.2. Real plant data analysis

In the above simulation study, it was shown that the DR estimation method, based on the UAR modeling by the Burg algorithm and the EBW relation, was most effective. Here, this algorithm is applied to real plant noise data and their practicality is examined. The noise data were obtained from a 1100MW e BWR/5 plant.

First, the estimated DR dependence on the sampling interval and UAR model order is shown in Fig.6. This figure shows that the estimated DRs are constant in the area of 0.1-0.5 second for the sampling interval and in the area of 15-30 for the model order. The order determined by the AIC seems to be adequate.

Next, results determined by the UAR and MAR modeling are discussed. Figure 7 shows PSD and DR calculated from the UAR model. The same DR can also be obtained by the MAR modeling. In Figs.8(a) and (b), both open-loop and closed-loop TFs, calculated by the MAR model, are shown. In these figures, it is seen that the DRs by the UAR model and by the closed-loop TF of the MAR model are in good agreement. As mentioned before, the closed-loop DR is a more practical index for stability monitoring. In this sense, a distinction between open-loop and closed-loop DRs is important, if MAR modeling is used.

Finally, the DRs are estimated in several operating points along the 100% power-flow control line and the 75% line. These DRs are compared with the analytical model predictions in Fig.9. Analytical predictions agree well with the measured data. The fact that both DRs decrease as the power increases along the control line is very plausible.

The above various examinations suggest the effectiveness of the stability estimation method from neutron noise data. However, they do not assure the absolute accuracy of the noise analysis method. To confirm this point, one of the authors compared the DRs estimated from noise data with those determined from the pressure perturbation test data (Kanemoto 1984). The results were satisfactory to assure that the noise analysis method was quantitatively reliable.

Fig.7 Power spectral density of APRM signal by the UAR modeling.

Fig.8 Transfer functions of APRM to core pressure drop by the MAR modeling.

Fig.9 Comparison of measured decay ratios with analytical predictions.
4. DEVELOPMENT OF ON-LINE STABILITY MONITORING SYSTEM

On the basis of the above examination, a prototype of the on-line stability monitoring system was developed. The objectives of the system are

1) To evaluate the reactor stability margin through DR estimation,
2) To detect homogeneous and inhomogeneous power oscillation in the core,
3) To supply process noise data acquisition and analysis tools.

In order to achieve these objectives, the system has the following software functions.

1) Real time UAR modeling and decay ratio estimation from neutron noise signals.
2) Real time monitoring of peak-to-peak (PKP) values of neutron noise signals.
3) Real time monitoring of a higher mode ratio of resonance peaks in the PSD.
4) On-demand UAR/MAR modeling and analysis,
5) FFT analysis,
6) Automated noise data acquisition.

Item 1) evaluation, which is based on the UAR model and the EBW relation, was carried out every 20 seconds and the results displayed. In this function, the time constant which corresponds to the noise data length can be arbitrarily specified. Item 2) detection was executed every 5 seconds. If the PKP values successively exceed the threshold at certain times, it can be regarded that power oscillation begins. Item 3) is useful to detect inhomogeneous power oscillation in the core, since such oscillation causes the higher mode oscillation in the average power range monitor signals. Items 4) to 6) are used for general purpose signal processing.

An outline of the system is shown in Fig. 10. The system consists of a 16 bit micro computer with 20 Mbyte disk memory, a CRT display, a hardcopy printer, a MT handler and a 64 channel A/D converter. These hardware components are combined in a compact rack.

Figure 11 shows an example of CRT display. The trend chart for the DR and PKP values are displayed in the upper part. In the lower part, core map and PKP values for each local power range monitor signal are displayed. In the left-hand side, current operating condition (power and core flow rate) are shown. The display can be changed by operator demand to show other figures, such as PSDs, raw data chart, etc.

The system was installed in an operating BWR/5 plant in one fuel cycle period. During that period, especially, in startup and shutdown conditions, various kinds of noise data were measured and the stability margin is monitored continuously.
Through this experience, the various system functions were well confirmed. Figure 12 is an example of the DR estimation results with various time constants (data lengths) in the above field test. It was seen that the scattering in the estimated DR depends on the time constant, but that the bias on the DR does not depend on the time constant. It should also be noted that the data length is needed longer than 400 seconds, in order to suppress the DR scattering. In various operating conditions, through out the one year field test, the stability margin monitored by the system was kept under a low level.

5. CONCLUSION

Several reactor stability estimation methods were presented and their accuracy and robustness were quantitatively evaluated, using simulation and real plant data. Furthermore, on the basis of these examinations, the on-line reactor stability monitoring system was developed. Through field tests on the system in a real plant, the validity of the system was confirmed. Deep understanding of the reactor stability phenomena was another valuable result through the analysis of various kinds of noise data aquired in the field test.

REFERENCES


APPLICATION OF NOISE ANALYSIS TO STABILITY DETERMINATION OF A NATURAL CIRCULATION COOLED BWR


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Abstract - Experiments were performed on the Dodewaard natural circulation cooled BWR at different conditions. The absolute stability was determined by measuring system responses to control rod and steam flow valve steps. Changes in core stability were studied using the signal of an average power range monitor (APRM) in time domain (auto-correlation function and impulse response) and in frequency domain (power spectral density and peaking factor), the outlet void fraction and variations of the incore coolant velocity.

It is shown that the reactor is very stable and that cooling by natural circulation improves load following. Stability monitoring can be performed by all mentioned methods but using APRM signals in frequency domain is preferred.

1. INTRODUCTION

Stability monitoring of a BWR is of great interest for safety aspects and for an efficient use of the reactor. Therefore the Dodewaard nuclear power plant has been subject to experiments concerning stability determination. An important feature of the Dodewaard BWR is that it is cooled by natural circulation.

Earlier measurements have shown some striking results on the stability of the reactor (Van der Hagen et al., 1986). By determining the reactivity-to-power transfer function (RTF) it was shown that the reactor kinetic stability, that has its origin in the reactivity feedback via the void fraction, is not significantly influenced by pressure changes. The neutron noise spectrum of an average power range monitor (APRM), however, did show some variation. This is due to variations of the input reactivity noise, that reflects thermal hydraulic stability.

Upadhyaya et al. (1982) have shown that the neutron noise spectrum gives indeed information on changes of the core stability. However, a perturbation test is needed for an absolute measurement of the stability.

The present paper deals with the application of methods and criteria that might be considered for monitoring the stability during operation.

Several ways of extracting information on stability from the neutron noise signal are in use:

a) in time domain
   - calculation of the auto-correlation function (March-Leuba and King, 1987)
   - calculation of the impulse response using an auto-regressive model (Upadhyaya et al., 1982).
   The decay ratio - defined as the ratio between two consecutive maxima - can be obtained from these functions.
b) in frequency domain

- fitting of the normalised auto power spectral density (NAPSD). Most authors use a second-order model (for instance Gialdi et al., 1985 and Pedercino and Ragona, 1986). March-Leuba (1986) proposes a functional form with 3 zero's and 4 poles as a model for the RTF; this indicates that a higher order model might be more successful.
- calculation of the root mean square (RMS) value over a certain frequency interval.

Changes of the spectrum that are linked up with changes of the stability can be observed on the basis of these findings.

Other possible stability criteria are the outlet void fraction of the hottest channel (presently in use for the Dodewaard BWR) and the standard deviation of the core coolant velocity.

Experiments were performed on the Dodewaard nuclear power plant in order to examine the validity of the above mentioned criteria. The absolute stability of the reactor was observed by measuring system responses to control rod and steam flow control valve steps.

2. EXPERIMENTAL CONDITIONS

Stability measurements were performed at 6 different experimental conditions, all of them at maximum power and nominal or lowered pressure. Table 1 lists the relevant process parameters.

<table>
<thead>
<tr>
<th>Exp.</th>
<th>Power (MWth)</th>
<th>Pressure (bar)</th>
<th>Cycle</th>
</tr>
</thead>
<tbody>
<tr>
<td>1</td>
<td>164</td>
<td>74</td>
<td>EOC 15</td>
</tr>
<tr>
<td>2</td>
<td>161</td>
<td>69.5</td>
<td>EOC 15</td>
</tr>
<tr>
<td>3</td>
<td>173.5</td>
<td>75.5</td>
<td>EOC 16</td>
</tr>
<tr>
<td>4</td>
<td>172</td>
<td>70.5</td>
<td>EOC 16</td>
</tr>
<tr>
<td>5</td>
<td>170.5</td>
<td>76</td>
<td>EOC 16</td>
</tr>
<tr>
<td>6</td>
<td>170.5</td>
<td>70</td>
<td>EOC 16</td>
</tr>
</tbody>
</table>

The pressure control was switched off during all the experiments. For experiments 5 and 6 the normal neutron flux profile was distorted by withdrawing the central control rods (all other rods were already withdrawn as the reactor was at the end of its cycle). In this manner a radial peaking factor as high as 1.02 was created at the centre of the core (normal value = 1.6). Thus the central channel was considerably less stable than at normal reactor conditions. The stability margin was even further reduced by lowering the reactor pressure at experiment 6.

Noise recordings of relevant signals at stationary conditions were performed during all experiments. During experiments 1 - 4 the responses of the system to control rod steps and to steam-flow control valve steps were also recorded.

3. STABILITY EVALUATIONS

3.1. Absolute stability

An indication of the absolute stability of the reactor was obtained by measuring the signal responses to control rod and steam flow control valve steps. A control rod step causes a local reactivity disturbance and enables thus the observation of the reactor kinetic stability. In order to obtain maximum reactivity effect the central control rod was chosen for, which was moved periodically between two positions in the core around the neutron flux maximum. The movement had an amplitude of 4 cm; its period was 4 minutes during experiments 1 and 2 and 8 minutes during 3 and 4. For a detailed description on the evaluation of rod and valve step responses during experiments 1 and 2 see Van der Hagen et al. (1986). Figure 1 displays the responses of the most relevant signals to a control rod insertion at the least stable situation at which step responses were determined, that is highest power and lowest pressure: experiment 4. The responses were averaged over 20 insertions in order to reduce the noise level. From this figure it is clear that good reactor kinetic stability is assured.

The steam flow control valve steps excite the core as a whole via the pressure reactivity effect. Therefore, the total system stability can be observed from the responses to these. Figure 2 gives the main averaged responses to a valve opening during experiment 4. The first 20-30 seconds correspond with a positive pressure-reactivity coefficient, as expected: pressure decrease leads to a higher void fraction which leads in its turn to moderator and thus
power decrease. Hereafter, the neutron flux increases to a level even higher than its initial value. This surprising effect, that arises from cooling by natural circulation, improves load following.

![Graph showing various flows and pressures over time]

**Fig. 1.** Signal responses to control rod insertion at experiment 4. Vertical scale division values are shown.

![Graph showing various flows and pressures over time]

**Fig. 2.** Signal responses to valve opening at experiment 4. Vertical scale division values are shown.

The increase in power is due to the colder recirculation water that reaches the core and increases the reactivity in two ways: it reduces the void fraction and its density is higher. The time-lag of this cold recirculation water can be calculated using the coolant voluma reported in Table 2.

For nominal conditions the steam density is 35.6 kg/m\(^3\) and the water density 740 kg/m\(^3\) (Kleiss and Van Dam, 1983). The averaged void fraction is 0.3 in the core and 0.5 in the chimney. The resulting coolant weights are also presented in Table 2. Assuming a recirculation flow of 1400 kg/s one obtains a time-lag of 26000/1400 = 19 seconds. The measured time-lag is larger: this is due to the fact that the recirculation flow through the bypass is very low and has a significant effect on the core reactivity.

The response of the steam velocity in the core was measured using a twin self-powered thermal neutron detector. It was found that the velocity increases 20 seconds after a valve opening (Van der Hagen et al., 1986). A thermal hydraulic analysis will be performed in order to understand this velocity effect.
Table 2. Volumina and weight of the coolant in the reactor vessel.

<table>
<thead>
<tr>
<th></th>
<th>Volumina (m³)</th>
<th>Coolant weight (kg)</th>
</tr>
</thead>
<tbody>
<tr>
<td>reactor core (channels)</td>
<td>1.96</td>
<td>1040</td>
</tr>
<tr>
<td>bypass</td>
<td>2.37</td>
<td>1750</td>
</tr>
<tr>
<td>chimney</td>
<td>7.87</td>
<td>3050</td>
</tr>
<tr>
<td>downcomer above sparger</td>
<td>2.73</td>
<td>2020</td>
</tr>
<tr>
<td>downcomer below sparger</td>
<td>12.50</td>
<td>9230</td>
</tr>
<tr>
<td>lower plenum</td>
<td>12.00</td>
<td>8880</td>
</tr>
<tr>
<td>Total</td>
<td>39.40</td>
<td>26000</td>
</tr>
</tbody>
</table>

In conclusion it can be stated that the Dodewaard reactor, due to natural circulation, is more selfregulating than forced circulation cooled BWR's.

The above mentioned findings indicate good stability at experimental conditions 1-4.

3.2. Monitoring changes of the stability

In this section the before mentioned methods of determining changes of the core stability will be presented for the different experimental conditions. The methods in time domain will be considered first.

![Image](Fig. 3. Auto-correlation function of the signal of an APRM for experiment 1 (most stable situation investigated) and experiment 6 (least stable situation).)

Figure 3 shows the normalised auto-correlation function of the signal of an APRM for two experimental conditions: situation 1 (considered to be the most stable of the investigated situations) and situation 6 (the least stable situation). The functions show a damped oscillatory behaviour with a frequency of approximately 1.1 Hz. The best damping can be observed for situation 1, as expected. However, the functions show another behaviour than those observed by others (for instance March-Leuba, 1984 and 1987 and Fry et al., 1984): in this case the decay ratio can clearly not be found by simply calculating the quotient of two consecutive maxima. It can be concluded that the auto-correlation function reflects stability but its use for stability monitoring is limited.

The impulse response of the APRM signal can be estimated from the AR model using an initial value response of the form

\[ h_i = \sum_{k=1}^{P} A_k h_{i-k} \]  

with initial conditions either

\[ h_0 = 1 \text{ and } h_{-k} = 0 \]  

(Fry et al., 1984 and March-Leuba, 1984)  

or

\[ \text{or} \]
\[ h_1 = 1 \text{ and } h_0, h_{-k} = 0 \quad (\text{Upadhyaya and Kitamura, 1981}). \] (3)

where \( h_1 \) = impulse response  
\( p \) = AR model order  
\( A_k \) = AR parameters

The actual derived impulse response depends strongly on the chosen initial values as the response at \( t=0 \) is forced to be either 1 or 0. The choice depends on the system in question; using conditions (2) lacks physical reality at \( t=0 \), as it implies an instantaneous response.

Fig. 4. Impulse response using initial conditions (2) derived by AR modelling of the APRM signal. The third-order fits are also shown (*).

Figure 4 gives the impulse responses for experiment 1 and 6 following Fry et al. and March-Leuba. The initial values mentioned by Upadhyaya and Kitamura were used for Figure 5. It is clear that the system under consideration is not of second-order type as the oscillatory behaviour is superimposed on an exponential decay. This leads to at least a third-order system. The first 3.5 seconds of the impulse responses using conditions (2) were fitted to the third-order model

\[ h(t) = A \exp(-\zeta \omega_o t) \sin(2\pi f_o \sqrt{1-\zeta^2} t) + B \exp(-t/\tau) \] (4)

with  
\( t \) = time  
\( \zeta \) = damping constant  
\( f_o \) = characteristic frequency  
\( \tau \) = time constant

by using the program 'FATAL', that uses a combination of Newton-Raphson, Steepest Descent and Marguardt algorithms for minimizing a sum of squares (Salmon and Booker, 1972). It can
be seen from figure 4 that the impulse response corresponding with the least stable situation is fitted best; a similar result was mentioned by Fry et al. (1984), who performed second-order fits.

![Impulse response using initial conditions (3)](image)

Fig. 5. Impulse response using initial conditions (3) derived by AR modelling of the APRM signal.

The decay ratio (DR) can be calculated from the model according to

$$DR = \exp(-2\pi T/\sqrt{1-T^2})$$

(5)

The resulting decay ratios and resonance frequencies $f_R = f\sqrt{1-T^2}$, are reported in Table 3.

<table>
<thead>
<tr>
<th>Exp.</th>
<th>Power(MWth)</th>
<th>Pressure(bar)</th>
<th>impulse response</th>
<th>NAPSD</th>
</tr>
</thead>
<tbody>
<tr>
<td>1</td>
<td>164</td>
<td>74</td>
<td>0.164 0.92 0.170 1.01</td>
<td></td>
</tr>
<tr>
<td>2</td>
<td>161</td>
<td>69.5</td>
<td>0.233 0.97 0.205 1.05</td>
<td></td>
</tr>
<tr>
<td>3</td>
<td>173.5</td>
<td>75.5</td>
<td>0.197 0.96 0.213 1.09</td>
<td></td>
</tr>
<tr>
<td>4</td>
<td>172</td>
<td>70.5</td>
<td>0.257 1.00 0.261 1.13</td>
<td></td>
</tr>
<tr>
<td>5</td>
<td>170.5</td>
<td>76</td>
<td>0.238 1.03 0.229 1.14</td>
<td></td>
</tr>
<tr>
<td>6</td>
<td>170.5</td>
<td>70</td>
<td>0.317 1.06 0.306 1.14</td>
<td></td>
</tr>
</tbody>
</table>

Table 3. Decay ratios and resonance frequencies obtained by fitting the impulse response and the NAPSD.

From this table it can be seen that a decrease in pressure leads to a higher decay ratio (worse stability) as does a higher power. Also the power distribution is of influence as can be seen from the decay ratios for experiment 5 and 6, where the strong flux peaking leads to a high value.

In spite of the good final results, an on-line determination of the core stability using the above mentioned method is strongly impeded by the computer work involved: data acquisition, spectrum determination, AR modelling, calculation of the impulse response, fit of the impulse response, calculation of the decay ratio. An extra problem is the fact that only the asymptotic decay ratio reflects core stability (March-Leuba, 1984). This ratio can differ considerably from the apparent decay ratio.

The second part to be dealt with is stability determination using the APRM signal in frequency domain. Changes of the stability influence the NAPSD in the frequency range around the break-frequency of the global neutron noise component (Upadhyaya and Kitamura, 1981). For the Dodewaard reactor this break-frequency is 1.1 Hz (Van der Veer, 1981). Therefore the spectrum was fitted from 0.65-2.0 Hz, using the inverse variance of the NAPSD, given by

$$\frac{1}{\sigma^2} = \frac{N}{\text{NAPSD}(f)}$$

(6)

($\sigma^2$ denotes the variance and $N$ the number of records used) as a weighting factor. The used model corresponds with the model used for fitting the impulse response.
NAPSD\( (t) = \left| \frac{A'}{1 - \frac{t^2}{f_o^2} + 2\imath \frac{t^2}{f_o^2}} + \frac{B'}{1 + \imath 2\pi f} \right|^2 \)  \( \text{(7)} \)

Figure 6 presents the NAPSD and its fit for situation 1 and 6.

![Graph showing NAPSD and fits for different experiments.](image)

**Fig. 6.** The NAPSD of the APRM signal and a third-order fit for the two extreme situations.

From the fits the decay ratio can be computed, using Eq. (3). The result can be found in Table 3. The determined decay ratios correspond well with those derived from the impulse response; the resonance frequency, however, is invariably higher. This is due to the fact that the oscillation frequency from the impulse response increases with time (Fig. 4). This points to a higher model order; the present model, however, suits our purposes.

![Graph showing peaking factor vs. decay ratio.](image)

**Fig. 7.** The peaking factor (the ratio of the RMS value of the APRM signal of 0.8-1.3 and 0.4-0.8 Hz) as a function of the decay ratio derived by spectrum fitting.
The RMS value of the APRM signal is formed by the noise source strength times the reactivity-to-power transfer function (both frequency dependent), integrated over a certain frequency interval. As this value can be easily computed it is obvious to try to extract information on the core stability from it. Figure 6 clearly shows that the RMS value (regardless of the frequency range) cannot be representative for the core stability; it is the relative height of the 1.1 Hz-peak that reflects stability. This peaking factor can be expressed as the quotient of the RMS value from 0.8-1.3 Hz and 0.4-0.8 Hz. These frequency ranges are chosen rather arbitrarily; the RMS value from the first interval is supposed to be influenced by stability changes, whereas the latter serves as a normalization value. Figure 7 displays this quotient - the peaking factor - as a function of the decay ratio derived by the spectrum fitting. It can be seen that the relation is remarkably linear for the investigated experimental conditions. Thus, the peaking factor is a convenient stability criterion for situations as those.

N.B. Also the signal of an incore neutron detector was investigated by the methods described above. Due to its large field of view around 1.1 Hz (Kleiss, 1983) the results were identical to those using APRM signals.

At present, three criteria regarding safety aspects are in use for the Dodewaard reactor:

a) the outlet void fraction of the hottest channel < 70%
b) the maximum heat flux < 136 W/cm²
c) the ratio of the minimum heat flux for film-boiling and the actual heat flux > 1.5

These criteria hold for 15% overpower.

In practice, only the first criterion limits the operating power. The outlet void fraction is calculated by the process-computer of the power plant. A plot of this fraction as a function of the decay ratio derived by spectrum fitting shows that the relation is linear (Fig. 8) and that it can thus be used as a stability criterion. However, an upper limit of 70% seems conservative regarding the system responses evaluated in section 3. Moreover, it is clear that the outlet void fraction criterion does not account for changes of the axial void fraction profile, that can influence the core stability. In fact the total void fraction of the hottest channel is more likely to be reliable.

![Fig. 8. The maximum outlet void fraction as a function of the decay ratio derived by spectrum fitting.](image)

As thermal hydraulic instability is characterized by large fluctuations of the coolant flow - especially in natural circulation cooled BWR's - variations of the incore coolant velocity were measured by correlating the noise of two axially displaced incore neutron detectors. For a description of the used detectors see Kleiss and Van Dam (1980). It turned out to be possible to estimate the steam-velocity with a standard deviation of 3.0% within 1.3 seconds (Van der Hagen and Hoogenboom, 1988). The standard deviation of the estimated velocity was 3.3% for situation 3 and 4.0% at situation 4. Thus hardly any significant velocity variations were found, which points to a stable reactor. The standard deviation at situation 4 is a bit higher than at 3 corresponding with the lower stability.

It is clear that this method - at the investigated conditions - is not sensitive enough to monitor core stability. An advantage at less stable situations is that it is the only method by which separate channel stability can be observed.
4. CONCLUDING REMARKS

System responses to control rod and steam flow valve steps indicate a good kinetic and total stability. Cooling by natural circulation improves load following.

Variations of the coolant velocity are too small to serve as a criterion for stability monitoring. Determining stability changes by calculation of the outlet void fraction of the hottest channel corresponds with monitoring using APRM signals, but it is felt that the first method is not always reliable and that the maximum allowable fraction of 70% (presently in use for the Dodewaard BWR) is conservative. The use of APRM signals is - in this reactor - complicated by the fact that the system cannot be described by a second-order model, as is usually done. Results using a third-order model are in close agreement. Using the peaking factor (the ratio between the RMS value of 0.8-1.3 and 0.4-0.8 Hz) is a convenient and reliable method for the investigated conditions.

REFERENCES


DYNAMIC CHARACTERIZATION OF THE BWR CORE/CHANNEL FOR THE CAORSO NUCLEAR POWER PLANT WITH APPLICATIONS TO STABILITY PATTERNS EVALUATION FROM OPERATING DATA

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ABSTRACT - The parallel growth of the technologies in transducers, measurements methods, digital signal processing and computing devices is the framework for the actual trend toward the development of built-in surveillance and monitoring systems that, from the incoming operational data, can infer the status of a plant and indicate significant deviations from normality. The request for such systems grows following safety and operational considerations.

In this work we present the bases on which ENEA, in collaboration with ANSALDO, has developed a local/global stability monitoring system for the Caorsu BWR plant; specific requirements have been posed in terms of real time and accuracy capabilities, so addressing the development of algorithms, specialized to the short-time stability margin estimation.

INTRODUCTION

Core and channel stability are important goals to be achieved in BWR design; neither any reactor channel nor reactor core should experience undesired oscillations of the flow and/or of the neutron flux.

Due to the non-linear nature of the BWR and to the voids negative feedback on neutronic power, the potential for unstable behaviour may exists; experimental tests conducted on BWR's have confirmed the possibility of limit cycles in the neutron flux, in the region of the operating map approximately at the crossing point between the natural circulation and the 100% rod line.

Depending mainly on the specific reactor power density, size and axial power shape, the limit cycle may appear in some different ways with the neutron flux that can oscillate either in-phase in all the core (global oscillations) or out-of phase (local oscillations).

The limit cycle conditions should be avoided and the situation of local oscillations must be considered with particular attention because in this case the average neutron flux (APRM) is no more representative of the real situation, showing oscillations of lower amplitude than the local ones.

Stability is not a problem during normal operations because, as stated before, the operating conditions at which the probability of uncontrolled oscillations is higher are not allowed and may be reached only in the unlikely event of a trip of both the recirculation pumps. In this case the operator is requested to insert as soon as possible control rods into the reactor core to achieve more stable conditions, that are considered to correspond to a power level equal to the 80% of the value on the 100% rod line at natural circulation.

Anyway the evidence of stable condition at this operating point may be discussed specially in the case in which local oscillations could occur; in any case the operator action must be fast enough to respect the plant operating limits.

Therefore an on-line stability monitoring system can be a valid tool for the operator during reactor operations in the boundary of the unstable region, if
real time response and specific information must be provided.
Of course the estimation accuracy and the real time capabilities do not always
live together in perfect harmony, but strong efforts in achieving these
performances must be dedicated in a monitoring system design because only in
this case the proposal is really alternative to the use of the technical
operating procedures.

1. BWR CORE AND THERMAL HYDRAULICS

The core of a BWR can be modelled as a multi-input/multi-output feedback
system in which the voids play a very important role on the reactor dynamics;
in fact a perturbation in voids changes the length of diffusion of the
neutrons, so changing the amount of reactivity in the reactor core. This
effect is described by the void coefficient that results negative for the BWR;
when the power increases a negative effect to the reactivity and to the
neutron flux verifies: the BWR is hence an inherently stable system.
Because of non-linearities and transport delays sometimes may happen that some
disturbance will be out-of-phase respect to the system input, producing an
amplification effect rather than a damped one. In this condition the limit
cycle is reached and the power must be decreased by the operator action.
Stability is generally represented by means of the Decay Ratio (D.R.) of the
impulse response of the system, supposed as a second order model; the system
is stable if D.R. is less than one.

Besides the core/channel stability problem, due to the coupling between the
neutronics and thermal hydraulics, another kind of instability should be
considered for the BWR: the hydrodynamic instability.
This kind of instability may happen in a channel in which a heated flow is in
saturated condition; a disturbance on the amount of heat transferred or on the
inlet flow can change the hydraulic resistance, then changing the pressure
loss and the energy stored in the steam and in the water. In this case self
sustained oscillations may arise because energy and continuity conservation
equations as well as momentum equations must be satisfied; this kind of
instability is observed without any nuclear effect as driving source.

Fig. 1 shows a schematic diagram of an experienced BWR dynamic model (Kanemoto
et al., 1984); the distribution of intrinsic noise sources, representing
stochastic fluctuations of the system variables, is there evidenced.
Detailed studies confirmed the model validity and stated that voids, core flow and pressure are the predominant noise sources contributing to the fundamental mode of oscillation of the neutron flux; the information on the so-called closed-loop stability can be derived from the dynamic relation between the neutron flux oscillations and the noise sources (Kanemoto et al., 1984). The closed-loop stability, because accounting for all the feedback effects showed in fig. 1, is considered more effective and realistic in the evaluation of the stability margin of the BWR; moreover its calculation can be approached with univariate modelling techniques.

2. DESIGN SPECIFICATIONS FOR THE BWR STABILITY MONITORING SYSTEM DESIGN

A significant reason of existence for a stability monitoring system is to consider it as an early stage diagnostic system of some undesired plant responses, to prevent the plant operations in the unstable region without any unnecessary penalty of the reactor power; moreover other particular needs from the utilities generally arise.

In any case the design specifications should have a real time content, so as the operator could promptly countermeasure in case of abnormal neutron flux oscillations without any exceeding of the plant operating limits.

The leading specifications for our system design were fixed in such a way to give the following minimum set of information (Della Casa et al., 1985):

- the Decay Ratio of the average neutron flux;
- the Decay Ratio of the local neutron flux on established core positions;
- an estimation error not exceeding 15% ;
- the peak-to-peak amplitude of both the local and average neutron flux signals;
- the phase between any two selected LPRM's;
- the resonance frequency of the fundamental mode of oscillation of the neutron flux;

The maximum time to present the information to the operator is to be close to 2 minutes including acquisition time, calculations and result presentation.

The above requirements were the bases for the design of the stability monitoring system, described in this paper.

3. THE METHOD FOR THE DECAY RATIO CALCULATION

The approach we followed in the identification of the BWR core/channel system refers to the univariate autoregressive modeling (UAR) methods. Design specifications justified the use of AR modelling because its capabilities in the representation of real systems over short observation time intervals are well known.

The basis of the AR modeling is that the signal can be considered as the output of a dynamic linear stationary system, fed by white noise source. Of course this hypothesis might be far from the real situation.

The effectiveness of such hypothesis in the BWR core/channel system has been demonstrated elsewhere (Kanemoto et al., 1984); as a result we can consider the neutron flux signal as the output of a system that can be described by a UAR model with a white noise input which all the BWR noise sources (voids, pressure, core flow) contribute to.

Deviations from this assumption may bias the results from the true values but our experience from the analysis of data confirms the reason of the choice.

The UAR model is defined as follows:

\[ y(k) = a_1 y(k-1) + a_2 y(k-2) + \ldots + a_n y(k-n) + e(k); \]  \hspace{1cm} (1)

In our procedure the model coefficients \(a_i\) are calculated by means of a least-squares procedure that, based on our experience, seems to furnish good performance in short time analyses.

The Decay Ratio estimation follows the calculation of the UAR model coefficients, using two parallel independent paths, starting from the identified transfer function of the system (see fig. 2); there \(H(z)\) is the transfer function of neutron flux to white noise.

The interest in doing multifunction methods is evident because the warranty in
continuity and reliability of calculation is greater when some of the estimates fails or yields large deviations from the actual trend; heuristic rules should be considered in the proper estimate selection.

![Diagram](image)

**Fig. 2** - Multifunction procedure for DR estimation

The first procedure, that is the calculation of D.R. based on the |H(w)|, fits a second order model:

\[
H(w) = \frac{a + j bw}{1 - c w^2 + j dw}
\]

to the estimated |H(w)| in a proper frequency band centered around the typical value of the stability resonance peak. The fitting is performed with a non-linear least-squares method; the leading criterion is to minimize the functional J, defined as:

\[
J = \sum_{i=1}^{M} \left( \frac{a + j bw_i}{1 - c w_i^2 + j dw_i} - H(w_i) \right)^2
\]

An iterative procedure to minimize J is followed in two steps to gain computational time and to assure convergence: firstly an initial guess close to the optimum solution is found with a low cost computational method; secondly a refinement step follows, starting from the previous estimate. This composite procedure gave successful and satisfying results in our applications.

A more simplified procedure is followed when the phase of H(w) is considered; in fact the phase slope at the resonance frequency depends on the damping ratio, and then on the Decay Ratio of the system assumed as before as a second order one.

The second order hypothesis is of conceptual nature and allows the measurement of the system stability margin in terms of the D.R. parameter.

The two methods have different ranges of applicability and accuracy; in particular it is more opportune to give emphasis to the phase estimation, when the resonance peak is not enough enhanced; in this case the linearity region of the phase is well defined. On the other hand the modulus procedure furnishes the best results when the resonance peak is higher.

The final estimation of D.R. arises from a weighted sum of the results of the two above described procedures where the weights are defined depending on the amplitude of the resonance peak. Fig. 3 and 4 show results from the two methods over the same signal.

4. **EXPERIMENTAL RESULTS**

The method for D.R. calculation above described was tested with artificial data as well as with experimental data.

4.1. **Artificial data tests**

Artificial data were built up to evaluate the accuracy of the procedure.
Fig. 3 - The DR estimation on the H(w) modulus.

Fig. 4 - The DR estimation on the H(w) phase.
Given an one zero/two poles analog system:

\[
H(s) = \frac{s + z0}{s**2 + 2pw + w**2}
\]  \hspace{1cm} (2)

associated in our approach to the stability phenomenon, several ways can lead to an equivalent digital system; one of those is the so called bilinear transformation (Papoulis, 1980) able to transform the (2) into the following:

\[
H(z) = \frac{Az**2 + Bz**1 + C}{Dz**2 + Ez**1 + F}
\]  \hspace{1cm} (3)

the free parameters (A,B,C,D,E and F) in (3) are defined by the corresponding parameters (z0,p and w) of (2) via the bilinear transformation.

A digital white sequence feeding the system (3) will cause an output with a spectral shape similar to that of the system that elaborates it.

In particular, sequences of any length, with an associated and chosen D.R. (related univocally to the damping ratio p) can be generated, suitably fixing the parameters of (3); it is what we did to build up the artificial data.

Twelve sequences were generated, each one 21600 samples wide, respectively with a D.R. ranging from 0.1 to 0.95. Table 1 gives an overview of the faced situations.

<table>
<thead>
<tr>
<th>RUN</th>
<th>1</th>
<th>2</th>
<th>3</th>
<th>4</th>
<th>5</th>
<th>6</th>
<th>7</th>
<th>8</th>
<th>9</th>
<th>10</th>
<th>11</th>
<th>12</th>
</tr>
</thead>
<tbody>
<tr>
<td>D.R.</td>
<td>0.1</td>
<td>0.2</td>
<td>0.3</td>
<td>0.4</td>
<td>0.5</td>
<td>0.6</td>
<td>0.7</td>
<td>0.75</td>
<td>0.8</td>
<td>0.85</td>
<td>0.9</td>
<td>0.95</td>
</tr>
</tbody>
</table>

Each of the generated sequences was broken up in consecutive windows 1200 samples wide; the corresponding D.R.'s were calculated with the above described method, obtaining 18 estimated values for each sequence.

Fig. 5 shows the results of these analyses; there only the mean value of D.R. is represented as well as the variance of the 18 estimates.

The following considerations can be done:

a) the method slightly underestimates the D.R. for medium/high values;
b) the variance of the estimate decreases as the D.R. increases.

The result in a) may be fully understood if a detailed analysis of the estimation procedure is conducted. In fact the estimate on \(|H(w)|\) (see par. 3) develops into two complementary steps, in which the first one finds out a first guess solution.

The method adopted in the first step is simple but has a drawback in the introduction of a negative bias that the next refinement step does not completely correct; in any case the bias is acceptable.

The result from b) is very interesting because demonstrates that the method has a greater accuracy at higher D.R. values.

4.2. Experimental data from Caorso tests

Stability tests were conducted on October 1983 at Caorso nuclear power station; the plant has a 2651 MWe GE-BWR/4 reactor. The limit cycle was reached at about 52% power and natural circulation; other tests were conducted in different operating conditions, in which the system pressure was perturbed changing the position of one of the main turbine control valves by a PRBS signal. Two different kinds of data acquisitions for each selected test condition were done: one at normal steady state, the other one during the pressure perturbation.

In this way a comparison between measurements obtained from two different
conditions, normal reactor noise and stimulated response, can be done. The PRBS perturbed conditions were analyzed with the classic Fourier Transform methods (Tricoli, 1985).

![Graph 1](image1.png)  ![Graph 2](image2.png)

**Fig. 5 -** Mean value $\overline{\text{DR}}$ versus generating DR.

**Fig. 6 -** Mean value $\overline{\text{DR}}$ versus PRBS DR.

The steady state conditions were analyzed (tab. 2); for each test condition, 24 minutes long, 12 windows (1200 samples, frequency rate = 10 Hz) were obtained and investigated. The mean values from the 12 estimates of D.R. are presented in fig. 6 versus the corresponding estimates from PRBS conditions. Interesting results are obtained:

- the correlation coefficient is very high ($r=0.9967$);
- the regression line slope is about 43 degrees;
- the intercept of the line with the $y$-axis is different from zero.

The first two results show deep coherence between the two sets of measurements, while the last item says that there is a mismatch between them, with the steady state D.R. lower than the PRBS D.R..

<table>
<thead>
<tr>
<th></th>
<th>TEST CONDITION</th>
</tr>
</thead>
<tbody>
<tr>
<td></td>
<td>1</td>
</tr>
<tr>
<td>D.R. s-s</td>
<td>.2314</td>
</tr>
<tr>
<td>D.R. PRBS</td>
<td>.3620</td>
</tr>
</tbody>
</table>

Table 2. - D.R. from steady state and PRBS data

Explanation of this might come from the already observed underestimation property of our method and from the overestimation property of the D.R. calculated from the classic method that refers to a reactor system model, in which the positive feedback of the neutron flux on the pressure is not considered (Della Casa et al., 1985).

4.3. The degrees of freedom of the procedure

The free parameters of the procedure are the sampling time, the length of the window and the model order. Sensitivity analyses of these parameters were done; considerations and results are presented below.
4.3.1. **Sampling time.** The definition of this parameter must satisfy different constraints; first the fundamental requirement from the sampling theorem, second the capabilities of identifying the variables of interest (Goodwin and Payne, 1977).

Our situation is not critical from the last point of view. A sampling rate of 10 Hz was found to be acceptable because several tests demonstrated that the stability resonance peak does never move its high side lobes over the 1.5 Hz value.

4.3.2. **Length of the analysis window.** The stability phenomenon is slow because relative to low frequencies.

A statistical constraint requires the availability of a sufficient amount of measurement points to assure an acceptable estimate of the parameter. On the other hand the need for a real time computation tries to minimize the number of the measurement points. An experienced trade-off led to the choice of a 1200 samples wide window, corresponding to two minutes of acquisition.

4.3.3. **Model order.** Objective criteria for A.R. model exist based on considerations about economics and significance and are used to define a range in which the optimum model order is to be found.

Generally other needs can arise; in our case the real time design requirements ask for the lowest model order without any detriment for the required accuracy, so also in this case experienced trade-off considerations led to a model order of 20.

5. **OTHER DYNAMIC INDICATORS OF THE SYSTEM STABILITY**

Another parameter generally used for the dynamic characterization of the system stability is the peak-to-peak amplitude of the neutron flux oscillations; while the D.R. is a shape factor of the system, the peak-to-peak is an amplitude indicator. The idea to relate D.R. to peak-to-peak is supported by the following two considerations:

a) the peak-to-peak amplitude of the neutron flux oscillation increases when moving toward the limit cycle condition;

b) the simplicity and the low computational cost of calculation of this parameter plays a positive role for the selection of such an indicator.

However specific investigations applied to Caorso data (Federico and Ragona, 1986) showed that this indicator cannot be used for an early detection of system instability.

In fact the fig. 7 presents the peak-to-peak versus D.R. for three Caorso test conditions, where both the coordinates are to be intended as the mean values among the estimated values (as usual, the signals were broken up and analyzed over consecutive windows).

The diagram among the calculated points seems to follow a step law, as represented in fig. 7 with a dotted line. This behaviour was also confirmed from the analysis of a slow transient at natural circulation, from 46.3% to 52.5% power.

These experimental circumstances give the peak-to-peak value the meaning of an auxiliary dynamic indicator, leaving the D.R. as the main representative of the stability margin.

6. **CONSIDERATIONS AND COMPARISON WITH OTHER DR ESTIMATION APPROACHES**

Several approaches have been tried in order to solve the problem of the BWR core/channel dynamic characterization; fundamentally we can distinguish between two different research lines: the former refers to the frequency domain approach, the latter to the time domain.

A method related to the frequency domain consisted in a pole-zero pattern identification, using the ARMA modelling of the neutron noise (Wu et al., 1981). This method suffers from a fundamental drawback, that rarely dominant poles can be well defined, from which to obtain the system dynamic characteristics; instead it happens often that poles aggregate in well defined clusters: in this case the definition of simple dynamic indicators (DR or
others) is by no means simple. Other methods referring to the time domain have been developed; the efforts were directed toward the system impulse response or the autocorrelation function analysis, that carry out informations about the dynamic status. The main difficulty here comes from the frequent presence of other dynamic modes near the stability resonance that affect the impulse response or the autocorrelation behaviour, requiring a proper processing of them; moreover, if non parametric methods are used to define such functions, large times are necessary in order to have significant and stable results.

In the last years an integrated procedure was proposed, with a combination of different methods (March-Leuba and Smith, 1984); here heuristic rules are set to decide among the estimated DRs.

The context for our method is the frequency domain; real time constraints (see par. 2) addressed the development of algorithms with short-time stability margin estimation and robustness capabilities. Heuristic rules are also applied (see par. 3) to average between the two estimation lines together with consistency checks. In our opinion other qualifying points are the ability to overcome the convergence difficulties inherent to least-squares fitting with the delineated two-steps procedure, and the faculty to integrate an a priori information that facilitates the next phases of estimation and result validation. We refer for example to the fact that the second order fitting is automatically conducted over the suitable frequency range (around 0.4 Hz for the stability phenomenon, so giving the result an intrinsic validation capability with respect to interferences from other dynamic modes.

7. THE PROTOTYPE
ENEa in collaboration with ANSALDO developed a stability monitoring system (Della Casa et al., 1985) following a detailed design path: setting up special hardware and electronics for data acquisition, developing scientific and graphic software and qualifying the code versus the Caorso stability tests.
the field test on Caorso plant has been scheduled for the next refueling.

Specific requirements have been accomplished in terms of local/global D.R. calculation capabilities; such need clearly arose after the analysis of the core oscillating behaviour for the Caorso limit cycle, that demonstrated the presence of local instabilities propagating to the remaining part of the core. Moreover the same analysis showed other interesting facts; first the core demonstrated to have an out-of-phase oscillation at about two opposite halves when in the limit cycle. Second a plane surface approximated to a good extent the neutron flux envelope of a fixed core height level; third the neutral axes positions of such planes resulted aligned in spatial sense (a dependence on some physical plant conditions was also supposed) and last that the plane surfaces oscillated with relative phases different from zero (Federico and Innarella, 1986).

At this moment the monitoring system is capable of tracking the D.R. values over the averaged (APRM) and a specified local (LPRM) neutron flux signal with a refreshing rate of one estimate every 2 minutes (due to accuracy requirement); in the same time the peak-to-peak values over the two signals are calculated, with a refreshing rate 10 times greater (one estimate every 12 seconds).

Further performances regard the capabilities of signal time history representation, of a complete frequency description (poles, zeros) and of a memory of the past estimates for validation of the actual ones; in fact the system holds memory of last estimated DR values and performs consistency checks against the current ones.

7.1. The hardware system

The data acquisition system was developed specifically for the stability monitoring system with the aim to solve particular problems related to the neutron flux signals measurement from the LPRM's. An opportune channel electronics was developed with the following main features (Belcredi and Giorgetti, 1986):

- large capabilities in amplification;
- high galvanic insulation;
- automatic offset compensation;
- digital filtering and undersampling capabilities;
- simultaneous sampling.

Each channel is equipped with a microprocessor for the handling of tasks related to conditioning and A/D conversion. Another electronic module synchronizes the different channel cards and transfers acquired data to the central CPU (microVAX II) via DMA. The data acquisition system can support 128 channels. The microVAX II takes the sampled data, provides to their elaboration and reduction, gives instructions to the graphic system in the interfacing with the operator. The maximum time to accomplish all these functions and give the final D.R. takes about 3 seconds for a channel.

7.2. The software system

The software system controls the data acquisition, performs the algebraic elaborations and produces graphics and messages to the operator. The system was implemented as a multitasking one, with several tasks contemporaneously activated in the central processor, all of them synchronized to perform a correct sequence of actions, and exchanging data each other with structures of shared memory. The amount of engaged memory for this system is about 180 Kbytes (data + programs area).

A general view of the prototype is shown in fig. 8; the compact rack in which the microVAX also can take place for an easy moving, is evident.
8. CONCLUSION
We presented an automatic system developed to monitor the local/global stability margin of the BWR core; it has been designed and suited to the Caorso NPP, but few efforts are requested to personalize it to other plants. Its design was addressed by defined constraints; noteworthy is its real time and local monitoring requirements.
Further developments are under study; the AR modelling should be moved toward an adaptive basis, because interesting properties in terms of fast tracking capabilities are known.
Moreover, current availability of powerful digital signal processor, when installed in the channel cards with proper software resources, can reduce the central processor charge in the computational tasks; in the limit, each local processor could accomplish the DR monitoring of the related channel, for a complete plant status description.

REFERENCES
OPERATIONAL EXPERIENCE
(PART III)

Session chairman: R. Baeyens (Belgium)
SUMMARY OF THE SESSION

This session, dedicated to operational experience, shows that noise analysis techniques are useful and maybe in some cases the only means to verify the validity of the design of the nuclear power plant or to pinpoint malfunctions or failures. These two aspects are well illustrated in three of the four papers. The final one is dedicated to standardising efforts made in USA in the noise field.

In the first paper by Blomstrand et al., the flow stability was verified in the case of new fuel elements in Swedish and Finnish BWRs.

The paper by Eitschberger et al. deals with the explanation of power fluctuations that happened at the Leibstadt BWR by noise analysis. This work will probably lead to design modifications in the recirculation loop. The success of the application of noise techniques leads us to see in the near future more and more standards and recommendations addressed to the utilities in order to help them in diagnosing problems. This was illustrated in the third paper by Zigler.

The paper by Guitton and Puyal on the French experience shows the tremendous efforts being made in France to develop new tools which can be used directly by the operators. The long positive experience of France in this field will be efficiently used in expert systems that will be installed in the French nuclear power plants.
NOISE ANALYSIS OF CORE COOLANT CHANNEL FLOW SIGNALS, RECORDED IN SWEDISH AND FINNISH BWRs

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Abstract - Noise studies of boiling water reactor (BWR) cores are usually based on signals obtained from the neutron detectors, for the lack of other in-core instrumentation. From the neutron flux noise, information can be gained on integral core performance, notably core stability. A number of BWRs in the Nordic countries are also equipped with instrumentation for inlet flow monitoring of some of the core coolant channels. In these particular plants, the in-core performance of individual fuel assemblies can be studied during reactor operation, via the noise in the corresponding channel flows. Such investigations have been made jointly by ASEA-ATOM and its customers for nuclear fuel in Finland and Sweden, in order to gain operational experience of fuel supplied by ASEA-ATOM, with respect to channel flow stability. The results indicate that fuel assemblies of a new design, called SVEA, have superior flow stability characteristics, compared to fuel assemblies of the standard 8x8 lattice design.

1. INTRODUCTION

At SMORN IV, a paper by Blomstrand and Andersson (1983) was presented, which outlined how noise techniques were applied for studies of dynamic characteristics of operating Boiling Water Reactors (BWRs) in the Nordic countries, and also presented some results from this work. At that time, noise studies were made on a fairly comprehensive scale by the reactor supplier (ASEA-ATOM), chiefly in the context of plant commissioning. The last studies of this kind were made in 1983, when both the Forsmark 3 and Oskarshamn 3 BWRs were commissioned.

Since then, the noise activities have turned towards more specialised applications, aimed primarily at gaining operational experience of reload fuel produced by ASEA-ATOM. This is a matter of common interest to both the fuel supplier and the customer. Accordingly, these noise studies are conducted as cooperative ventures between the parties concerned: involving - apart from ASEA-ATOM - a majority of the power companies in Sweden and Finland that own and operate BWRs plants using reload fuel supplied by ASEA-ATOM.

For obvious reasons, these studies chiefly concern aspects that relate to core performance in operating BWRs, for example, stability. More specifically, it is of some interest to investigate what impact the insertion of reload fuel in a BWR core may have on stability. For reload fuel of new designs, it is desirable to acquire such operational experience as early as possible, i.e. as soon as the first few demonstration assemblies have been loaded into a reactor core. When noise techniques are to be used for gaining this information, these techniques must be applied in such a way as to permit performance studies of individual fuel elements in operating BWR cores.

a undergraduate from the Polytechnic of the South Bank, Department of Physical Sciences & Technology, London
Some of the BWRs in the Nordic countries are equipped with instrumentation for inlet flow monitoring of individual core coolant channels, permitting noise studies of channel flow signals. During the last years, such investigations have been made in Forsmark 2 & 3, TVO II and Oskarshamn 3, to provide information about the stability characteristics of coolant flows through fuel assemblies supplied by ASEA-ATOM. This paper outlines how these studies have been conducted, and presents some salient results obtained to date.

2. BWR STABILITY

A brief summary of the subject is given here, for the sake of completeness. In BWR cores, there are two stability phenomena of interest: core stability and channel flow stability.

Channel flow stability involves the thermal-hydraulic behaviour of a two-phase mixture flowing through a heated channel. From the kinetic point of view, the liquid content of the channel constitutes a "mass", whereas its vapour content forms an elastic "spring". It is well-known that a dynamic system comprising a mass and a spring may oscillate; consequently, the water-steam mixtures flowing through the coolant channels of a BWR core may show tendencies towards thermal-hydraulic flow fluctuations. The dynamic characteristics (natural frequency, damping, etc.) of these fluctuations depend on:

i) the geometry of the heated channel (its length, hydraulic diameter, heated perimeter, inlet orificing, etc.), and

ii) its operating conditions (mass flow, inlet sub-cooling, channel power, axial power distribution along the channel, etc.).

Thermal-hydraulic flow fluctuations in heated channels have a tendency to propagate as "density waves", i.e. travel along the channels at about the same velocity as the two-phase flow itself. For this reason, the natural frequencies of such flow fluctuations are related to the coolant transport times through the channels.

As regards possible coupling effects between flow fluctuations in individual channels through a BWR core - and the possible coupling of such flow fluctuations to the neutron flux - it should be appreciated that the coolant channels all have the same lengths, and operate under common boundary conditions (the same inlet sub-cooling, and the same driving pressure: the pressure difference between the lower and the upper core plenum). Otherwise, the channels experience different operating conditions, in terms of channel powers, axial power distributions, channel flows, etc. Different coolant channels may even contain fuel assemblies of different designs. Against this background, it is reasonable to expect that thermal-hydraulic flow fluctuations which occur within each individual channel should be governed by "local" conditions, i.e. pertaining solely to the channel considered. Accordingly, one may anticipate that the individual flows through the coolant channels of a BWR core should fluctuate independently of each other, in other words "incoherently". Channel flow fluctuations of this nature may reasonably be expected to cancel each other out, with regard to their net influence on reactivity - and thereby their response in the global neutron flux. Consequently, thermal-hydraulic channel flow fluctuations are unlikely to be affected by "nuclear feedback".

Core stability, on the other hand, is a phenomenon that involves significant nuclear feedback. An initiating event could be a global reactivity disturbance (which may have been triggered by an external disturbance, for example: a sudden fluctuation in reactor pressure), causing a global response in the neutron flux - and, thereby, in the fission rate (i.e. the heat generation rate in all the fuel rods). Such fluctuations in the heating of the nuclear fuel cause fluctuations in the heat fluxes on the fuel rod surfaces, to an extent - and with a time lag - that is determined by the heat transfer properties of the fuel rods. The fluctuations in the heat input to the coolant channels influence, in turn, their steam content - to an extent that depends on their characteristics, with respect to channel flow thermal-hydraulics. Since the original disturbance in heat generation rate of the fuel was coherent, one may expect all the coolant channels involved to respond in a coherent manner. In a BWR core, such coherent fluctuations in the steam content of its coolant channels combine to produce a significant reactivity response, via the "void-reactivity coupling" - thereby closing the "nuclear feed-back" loop back to the heat generation rate in the fuel. Because of the strong neutronic coupling between adjacent fuel assemblies in the core, the neutron flux fluctuations tend to extend over fairly large core regions: normally the entire core.

To summarise: in BWR cores, operating at power, two different kinds of fluctuations superimpose on the flows through all the coolant channels:

i) thermal-hydraulic flow fluctuations of a "local" nature, i.e. involving the individual coolant channels, and

ii) coupled fluctuations between channel flows and neutron fluxes of a "global" nature, i.e. extending over groups of coolant channels.

Since the "nuclear feedback" involved in the "global" flow fluctuations includes a time delay (that from heat generation in the fuel to the fuel surface heat flux), these fluctuations have different dynamic characteristics (i.e. natural frequency and damping), as compared to the "local" channel flow fluctuations. Yet, channel flow stability and core stability are not entirely unrelated phenomena, since they both involve channel flow thermal-hydraulics.
One of the stability requirements for BWRs is that possible flow fluctuations occurring in any of the core coolant channels should be reasonably well damped, for all normally permitted operating conditions. To satisfy this, the following measures are taken:

i) orificing the channel inlets, and

ii) ensuring - by administrative means (to be described below) - that the channels never experience excessive steam/water ratios.

3. IN-CORE INSTRUMENTATION

Over the years, a considerable number of noise studies have been made in BWRs around the world, with the intention of extracting information on core performance from the noise patterns seen in process signals obtained from the in-core instrumentation. These studies have almost exclusively been based on the neutron flux noise - since BWR cores are normally equipped quite generously with instrumentation for neutron flux monitoring (LPRM, APRM). In this way, several investigators have been able to obtain valuable information on core stability.

Studies of channel flow stability in operating BWRs via noise techniques are hampered by the fact that core coolant flows are usually monitored via sensors located in the external parts of the coolant recirculation circuit. Nevertheless, several investigators have successfully made use of the noise patterns seen in the signals obtained from the neutron sensitive in-core instrumentation, in order to extract interesting information on core coolant transport. However, since each LPRM detector tube is located in the gap between the four adjoining fuel elements, the LPRM signals present a smeared-out picture of what happens in the four adjacent coolant channels. Detailed events in individual channels cannot, therefore, be clearly resolved via the LPRM signals.

Measurements of individual channel flows through BWR cores have been made previously (e.g. in Halden and Oskarshamn 1), by means of special instrumentation. However, some of the Nordic BWRs are equipped with permanent instrumentation for inlet flow monitoring of individual core coolant channels. Attention to this was drawn already in the above-mentioned paper by Blomstrand and Andersson (1985). Six such BWRs are in operation:

i) TVO I and II in Finland, and

ii) Forsmark 1 - 3 and Oskarshamn 3 in Sweden.

A brief description of the flow monitoring instrumentation follows. It was mentioned above that every BWR core has an orifice at the inlet of each of its coolant channels, to ensure satisfactory stability characteristics. These orifices also serve to provide a reasonable flow distribution over the core. They are normally mounted in the fuel support plates. In each of the BWRs mentioned above, eight selected coolant channels are provided with pressure taps up- and downstream of their inlet orifices. From these taps, small-bore pipes are run in pairs, first to penetrations in the lower part of the reactor vessel, and then through the containment wall - on to transmitters for differential pressure measurements. The sensed pressure drop signals are first combined with the flow resistance coefficients of the precalibrated orifice plates, and then converted to channel flow signals via root-extracting devices.

As an example, fig. 1 presents the locations of the flow-monitored coolant channels in the cores of the Oskarshamn 3/Forsmark 3 reactors.

Among commercial BWRs, the instrumentation is unique. It was installed primarily to make it possible to determine the total core coolant flow accurately during steady-state operation, via precise measurements of the flows through a number of representative core coolant channels. The fact that the same instrumentation can also be utilised for observing the time-dependent behaviour of the channel inlet flows is an additional bonus. The instrumentation has such fast response characteristics that the flow signals are able to reproduce even quite rapid flow changes (at least in relation to the time-scale set by the natural frequency of the thermal-hydraulic flow fluctuations). Therefore, the noise patterns of the channel flow signals contain useful information about the stability characteristics of the individual coolant channel flows monitored, for example how they depend on channel operating conditions and fuel assembly design.
4. FUEL DESIGNS

Initially, the cores of all the Nordic BWRs were made up of fuel assemblies of the conventional 8x8 design, i.e. each bundle containing 64 rods in a square lattice, enveloped by a fuel shroud. The fuel assembly design is illustrated schematically in fig. 2 a.

About 1980, development work started at ASEA-ATOM on a new design for BWR fuel, called SVEA. The same type of fuel design is marketed by Westinghouse under the name QUAD+. There are currently two basic SVEA design versions: SVEA-64 and SVEA-100 - containing 64 and 100 fuel rods. The respective designs have been described in some detail by Nylund et al (1981), and by Nylund et al (1986). A brief summary is given here of their main design features.

SVEA-64 may be visualised as a conventional 8x8 fuel rod bundle subdivided into four "mini-bundles", each comprising 16 fuel rods in a 4x4 lattice. The SVEA assembly is built up of these four mini-bundles in a square lattice arrangement, with each mini-bundle inserted within its individual sub-channel. The sub-channels are joined together so that a clearance appears between them, forming a central space of cruciform shape in each fuel element. When the fuel element is submerged in water (for example in a BWR core), this central space is occupied by non-boiling water; for this reason it is often referred to as a "water cross". The chief virtue of the SVEA design relates to neutron physics: the water cross flattens the thermal neutron flux across the fuel assembly, improving the neutron economy. Fig. 2 b presents a schematic view of the SVEA-64 assembly design.

SVEA-100 is basically similar to SVEA-64, the main difference being that the 16 rod mini-bundles are replaced by mini-bundles containing 25 thinner fuel rods, in a 5x5 lattice. Fig. 2 c presents a schematic view of the SVEA-100 assembly.

![Fig. 2 a. 8x8 lattice](image1)

![Fig. 2 b. SVEA-64 lattice](image2)

![Fig. 2 c. SVEA-100 lattice](image3)

To improve the coolant flow distribution through the fuel assembly, the SVEA fuel shroud incorporates a number of refinements:

i) Barriers at the lateral ends of the "water cross" wings prevent the non-boiling water, flowing through the water cross, from mixing with the bypass water, flowing through the gaps between the fuel assemblies.

ii) To achieve pressure equilibration throughout the entire length of the assembly between adjoining sub-channels (as described above), communication openings are provided that permit some lateral mixing of coolant.

Apart from the neutron physics improvements, the SVEA design also improves performance with respect to channel flow stability - this was realised already in early loop experiments. Additional evidence has emerged from noise studies of channel flow signals, recorded in those of the BWRs mentioned above which have received SVEA fuel.

5. FLOW NOISE STUDIES

The first studies of channel flow noise in the BWRs equipped with instrumentation for channel inlet flow monitoring were made by ASEA-ATOM during commissioning of these plants. On these occasions, their cores contained only 8x8 lattice fuel.

In 1984, SVEA-64 demonstration assemblies were loaded for the first time into the core of such a BWR: Forsmark 2. One of these assemblies was loaded into a flow-monitored core location. The first noise studies including the channel flow through this particular assembly were made by personnel at Forsmark. The outcome was so interesting that channel flow noise in BWRs containing SVEA fuel have since then been studied in a systematic manner.
In 1985, more SVEA-64 demonstration assemblies were loaded into Forsmark 2 - and also into TVO II. In 1986, Forsmark 2 received an entire reload of SVEA-64 fuel, and SVEA-100 demonstration assemblies were loaded for the first time into BWR cores: both Forsmark 3 and Oskarshamn 3. On each of these occasions, care was taken to make sure that at least one SVEA fuel assembly was loaded into a flow-monitored core location.

The noise studies have concentrated on investigating:

i) the stability characteristics of the individual channel flows (through coolant channels containing 8x8 fuel, as well as SVEA fuel).

ii) possible coupling effects between the individual channel flows, and

iii) possible coupling effects between the individual channel flows and the neutron flux.

Accordingly, most of the noise recordings have comprised all the eight channel flow signals, plus one APRM signal (which normally contains sufficient information on the "global" fluctuations in the neutron flux). The majority of the recordings have utilised fairly high scanning rates (to obtain good quality of the raw data collected), in combination with a large number of scans per signal (to provide good statistical accuracy of the results).

The recording equipment has been described in the above-mentioned paper by Blomstrand and Andersson (1985), and the same applies to the methods employed for the data analysis. Briefly, the recordings are made by means of computerised data collection systems. One such system is installed permanently in each of the ASEA-ATOM BWRs equipped with instrumentation for channel flow monitoring. A large number of signals from the plant instrumentation are connected to this recording system on a permanent basis, so that they are available for recording at any time. The particular signals that are to be recorded in connection with any specific activity (for example, a recording of noise in channel flow signals) are selected by the operator of the data collecting system, prior to the recording. The recorded data are stored on discettes, which are subsequently analysed offline. The data reduction is based on

i) evaluations in the time domain for individual signals: qualification of raw data, and evaluations of average values, noise levels, amplitude distributions, auto correlations, etc., and

ii) evaluations in the frequency domain (utilising FFT techniques): calculations of auto spectra for individual signals, and cross spectra and transfer functions for signal pairs, including coherence analysis.

The flow noise studies are currently conducted according to the following procedure:

i) details of when and how the noise recordings should be made in a particular reactor are discussed in advance between ASEA-ATOM and the power plant personnel,

ii) the recordings are then made by the plant personnel,

iii) the discettes are forwarded to ASEA-ATOM for detailed analysis.

The analysis results are only meaningful in relation to the operating conditions of the flow-monitored channels. In this respect, the evaluations of the noise recordings provide time-averaged values for the channel flows concerned. The remaining information on channel powers and axial channel power distributions are obtained from reactor physics calculations with the core simulators. These are usually made by personnel in the plants concerned.

It has been our ambition to study channel flow noise in a variety of operating situations of interest. Some examples are described here, with reference to the "power/flow diagram" shown in fig. 3. The combinations of power and flow that are permitted for normal operation are indicated. Other power/flow combinations are forbidden, for the following reasons:

i) operation below the minimum permitted coolant flow (to avoid flow-starved coolant channels),

ii) operation above the maximum permitted coolant flow (to avoid excessive creep deformation of the fuel shrouds),

iii) operation above the maximum permitted power (to avoid exceeding the designed plant output), and

iv) operation at excessive power/flow ratios (for reasons connected with concerns about dryout and stability, as discussed above).
During an operating season between two successive refuellings, a BWR core normally experiences the following (somewhat over-simplified) history:

At startup, the coolant recirculation pumps are run at their minimum permitted speed, while core power is raised by pulling out control rods, until about 65 - 70 % power is reached (point A in fig. 3). From here, core power is raised by increasing the coolant flow, by means of the recirculation pumps. At full power, the plant is normally operated at the minimum permitted flow (point B in fig. 3), while the remaining control rods are gradually withdrawn from the core. During this time period, the need may arise for some temporary power reductions (for example, at weekends); these are achieved by reducing the coolant flow via the recirculation pumps (from point B in fig. 3 to point C, for example). When all the control rods have been withdrawn from the core, "stretchout" operation follows, whereby - at full power - the coolant flow is slowly increased until its upper permitted limit is reached (point D in fig. 3). "Coast-down" follows, whereby the core power slowly decreases (from point D in fig. 3 down to point E, for example). The plant is then shut down for refuelling.

![Fig. 3. Typical BWR operating map](image)

6. RESULTS

A selection of results obtained from the spectral analysis of the noise recordings are presented. All auto spectra are plotted in a linear frequency scale and a logarithmic amplitude scale; and both the calculated channel power factors and the noise levels experienced are quoted. The latter are expressed in terms of the standard deviations, divided by the corresponding time-averaged values. Each plot is individually scaled - this must be borne in mind when different auto spectra are compared.

Since the core usually operates at the minimum coolant flow permitted at full power (i.e. at point B in fig. 3), most of the noise recordings have been made under these operating conditions - although other operating points have also been investigated. In all the operating situations considered the stability margins were satisfactory.

6.1. Channel flow noise

6.1.1. Channel flows through 8x8 fuel assemblies.

Fig. 4 illustrates some general features of noise auto spectra in channel inlet flow signals recorded at the minimum coolant flow permitted at full power (point B in fig. 3). The results originate from TVO II, and refer to a coolant channel operated at a channel power somewhat higher than the core average - the calculated channel power factor (f-rad) = 1.20. In this example, three spectral peaks can be observed - a strong one at about 1 Hz, a weaker one at about 2.5 Hz, and a faint one at about 3.4 Hz. Apparently, the channel flow fluctuations involve at least three separate oscillatory modes. They all appear to be fairly well damped, according to the half-widths of the corresponding spectral peaks.

![Fig. 4. Typical channel flow noise auto spectrum](image)

The fundamental mode component accounts for most of the channel flow noise, whereas the third oscillatory mode is so relatively insignificant that frequencies above 3 Hz do not contribute appreciably to the noise. For this reason, the remainder of the results presented in this section are restricted to the frequency range 0 - 3 Hz, and all noise levels quoted in the figures below comprise contributions from spectral components within this range only.

Figs 5 a - c illustrate how auto spectra of channel flow noise depend on channel power. The results originate from a recording made in Oskarshamn 3 at the minimum coolant flow permitted at full power (i.e. the same operating point that was considered above).
In both figs 5 a and b (describing a fairly high-powered and a moderately powered channel, respectively) prominent spectral peaks appear at about 1 Hz (the same frequency as was seen in fig. 4), whereas in fig. 5 c (corresponding to a rather low-powered channel), only a very faint peak can be perceived at this frequency. The figures illustrate that the peak amplitude of the fundamental mode flow oscillations is strongly channel power dependent, increasing with higher channel power (the same also applies to the flow noise). On the other hand, the natural frequency of the fundamental mode component of the flow noise appears to be fairly independent on channel power. In this context, it may be appreciated that the flows through the three coolant channels considered were reasonably similar.

Shortly after this noise recording was made, there was a periodic test of the steam line isolation valves, which required a power reduction down to about 75%. This was achieved by lowering the core coolant flow (i.e. moving from point B in fig. 3 down to point C). Noise was recorded in this operating point. Figs 6 a - c present the noise auto spectra obtained, for the same channel flow signals that were considered in figs 5 a - c above.

Apparently, lowering both the channel powers and the channel flows in this manner reduced the natural frequencies of the fundamental mode oscillations, from about 1 Hz (in figs 5 a - b) to about 0.7 Hz (in figs 6 a - b), but increased significantly the amplitudes of the spectral peaks - and also the noise levels.
The questions arise, then: how are channel flow noise patterns affected by changes in channel flows at constant power, and, conversely, by changes in channel powers at constant flow? This may be illuminated by studying how channel flow noise patterns change during reactor operation at stretchout and coastdown.

Figs 7a and b illustrate how flow noise auto spectra are affected during stretchout (i.e. moving from point B to point D in fig. 3). The results originate from two noise recordings in Oskarshamn 3: one was made at the initiation of stretchout, at the minimum coolant flow permitted at full power (fig. 7a), the other one was made about one month later, at an almost 30% higher coolant flow (fig. 7b).

Apparently, the corresponding channel flow increase - at an essentially constant channel power - raised the natural frequency for the fundamental mode flow oscillation, from about 1 Hz to about 1.4 Hz, but reduced the amplitude of the corresponding spectral peak - and the flow noise.

Recall here that stretchout operation tilts the axial core power distribution upwards significantly - which affects the axial power distributions in all the coolant channels similarly. For this reason, the spectral differences between figs 7a and 7b do not merely reflect the influence of the increased channel flow, but also that of the increased up-ward tilt of the axial channel power distribution.

Figs 8a and b illustrate how flow noise auto spectra change during coastdown, i.e. from full power (at point D in fig. 3) down to reduced power (point E). The results originate from two recordings made in Forsmark 3 at the maximum permitted flow: one at 100% power (fig. 8a), and the other about one month later, at 36% power (fig. 8b). Apparently, reducing the channel power at constant coolant flow left the natural frequency of the fundamental mode flow oscillation essentially unchanged, but reduced the spectral peak amplitude, and thereby also the flow noise.

Summarising, the results presented in figs 5-8 suggest that as far as the fundamental mode component of the flow noise is concerned, the natural frequency depends mainly on the channel flow, whereas the peak amplitude depends on both the channel power and the channel flow, as well as on the axial power distribution. The highest flow noise occurs in high-rated coolant channels with downward-tilted power distributions, in operating situations with relatively high power/flow ratios (such as the points A, B and C in fig. 3).

In order to investigate to what extent noise patterns of channel flows through different 8x8 lattice assemblies are coupled to each other, a large number of signal pairs involving such channel flows have been cross analysed. The coherence patterns obtained all show the same behaviour: no significant coherence can be observed.

It was mentioned above that both incoherent (i.e. purely thermal-hydraulic) and coherent (i.e. neutron flux coupled) flow fluctuations contribute to the total channel flow noise. The observed lack of coherence between the noise patterns of the different channel flow signals indicates that as far as 8x8 lattice assemblies are concerned, the purely thermal-hydraulic flow fluctuations provide the dominant contribution to the total channel flow noise. This conclusion applies to all normal operating conditions (i.e. situations in which the core stability margins are satisfactory).
6.1.2. Channel flows through SVEA assemblies. Figs 8 a - c illustrate the fundamental difference between noise spectra of channel flows through SVEA-64 assemblies and 8x8 lattice assemblies. The results originate from a noise recording in Forsmark 2, made in an operating point at the minimum permitted coolant flow at full power (point B in fig. 3). Fig. 9 b presents the noise auto spectrum of a channel flow through a SVEA-64 fuel assembly operating at a fairly high channel power. This is to be compared auto auto spectra for flows through two coolant channels containing 8x8 lattice fuels: one (fig. 9 a) from a coolant channel having essentially the same channel power as that containing the SVEA-64 assembly, the other from a coolant channel having a lower power (fig. 9 c).

![Fig. 9 a. Flow noise auto spectrum, 8x8 lattice fuel](image1)

![Fig. 9 b. Flow noise auto spectrum, SVEA-64 lattice fuel](image2)

![Fig. 9 c. Flow noise auto spectrum, 8x8 lattice fuel](image3)

As expected, peaks appear at about 1 Hz in the auto spectra for the channel flows through both the 8x8 lattice assemblies (figs 8 a and c). No significant peak can be perceived at this frequency in the noise auto spectrum for the channel flow through the SVEA-64 assembly (fig. 8 b). This indicates that the coolant flow through the SVEA-64 assembly is more stable than the flows through either of the 8x8 lattice assemblies. By now, many noise recordings have been made of channel flows through several different SVEA-64 demonstration assemblies, in both Forsmark 2 and TVO II. A variety of operating points have been investigated. The experience from the evaluations of all these recordings agree with the findings of figs 9 a - c.

Noise recordings have so far been made of channel flows through two SVEA-100 assemblies, one in Forsmark 3 and one in Oskarshamn 3, in a variety of operating points. To illustrate the results obtained, figs 10 a - c shows auto spectra of channel flow noise, recorded in Oskarshamn 3 at the minimum permitted coolant flow at full power (point B in fig. 3). They are presented in the same manner as those in figs 9 a - c.

![Fig. 10 a. Flow noise auto spectrum, 8x8 lattice fuel](image4)

![Fig. 10 b. Flow noise auto spectrum, SVEA-100 lattice fuel](image5)

![Fig. 10 c. Flow noise auto spectrum, 8x8 lattice fuel](image6)

The noise auto spectra of the channel flows through the two high-powered fuel assemblies show the same basic differences between the 8x8 lattice fuel (fig. 10 a) and the SVEA-100 fuel (fig. 10 b) that were seen above in figs 8 a - b. Stabilitywise, the channel flow through the high-powered SVEA-100 assembly (fig. 10 b) behaves in much the same way as the flow through the low-powered 8x8 lattice assembly (fig. 10 c). The other flow noise recordings involving SVEA-100 assemblies show similar results; channel flows through SVEA-100 assemblies are more stable than flows through 8x8 lattice assemblies.
6.2. Coupling effects between channel flow noise and neutron flux noise

The results from the cross analyses reported under 6.1.1 indicated that as far as the channel flows through the 8x8 lattice assemblies were concerned, the flow noise was dominated by the incoherent (i.e. purely thermal-hydraulic) flow fluctuations. This implies that the coherent (i.e. neutron flux coupled) flow fluctuations constitute only a minor fraction of the total flow noise. The same may not necessarily apply to channel flows through SVEA fuel assemblies, in view of the lower flow noise. It is possible to obtain an improved picture of the magnitude of the neutron flux induced contribution to the total flow noise, by cross analyses of channel flow signals and neutron flux signals.

The noise recordings discussed in this paper have all been made while the reactor cores that were the objects of the noise studies still contained only a small number of SVEA demonstration assemblies. Under these circumstances, the stability characteristics of these cores were largely governed by their large inventories of 8x8 lattice fuel. All the operating situations investigated had satisfactory core stability margins.

The results presented in figs 11 a - e and 12 a - e refer to two noise recordings from TVO II. They both refer to the same operating point (the minimum coolant flow permitted at full power, i.e. point B in fig. 3), but were recorded while the plant power control system was operating in different control modes. Acting on the coolant flow via the recirculation pumps, this system strives to keep the plant output at a set point value.

Figs 11 a - e refer to noise recorded in a control mode where the control system was practically disconnected, i.e. all the coolant recirculation pumps were set to run at a constant speed. Fig. 11 c presents the corresponding auto spectrum for the APRM signal (which behaves in a typical manner, in the operating situation considered). Noise auto spectra from the same recording are also given for channel flows through two equally-powered fuel assemblies, one of the 8x8 lattice design (fig. 11 a), the other of the SVEA-64 design (fig. 11 e).

The peak appearing at about 0.75 Hz in the noise auto spectrum of the APRM signal (fig. 11 c) signifies that in the operating point considered, the neutron flux has a certain tendency to exhibit damped oscillations at this frequency: "the core resonance". This is to be compared with the noise auto spectrum of the channel flow through the 8x8 lattice assembly (fig. 11 a), where a peak appears at 1 Hz: the same frequency that was seen in figs 4 and 5. These results show that the coolant flows through 8x8 lattice assemblies fluctuate significantly faster than the neutron flux (again in the operating point considered). On the other hand, fig. 11 e indicates that the channel flow through the SVEA assembly shows no particular tendency to fluctuate at either of these frequencies - nor at any other frequency (in line with the experience from fig. 9 b).

Figs 11 b and d present the coherence patterns between the APRM signal and these two channel flow signals. The absence in fig. 11 b of any significant coupling between the signals (even at the frequencies where the peaks appear in the auto spectra of either of them) indicates that the channel flow through the 8x8 lattice assembly considered and the global neutron flux fluctuate independently of each other. This behaviour is typical for the coherence patterns of such signal pairs. Conversely, fig. 11 d shows that in the frequency interval around the core resonance, a weak coupling may be perceived between the APRM signal and the channel flow through the SVEA assembly. This behaviour should not be entirely unexpected. It was mentioned above that the core resonance phenomenon arises from an interplay between coolant flow thermal-hydraulics, neutron physics and fuel rod heat transfer. It was also pointed out that the fluctuations in the neutron flux at the core resonance frequency can be expected to excite coherent responses in all the channel flows - and this applies to channels containing 8x8 lattice fuel as well as SVEA fuel. These neutron flux induced flow fluctuations are superimposed on the purely thermal-hydraulic channel flow noise - which is much higher in the former channels than in the latter ones, according to the results presented above.
The neutron flux induced channel flow fluctuations may, therefore, be expected to occupy a larger fraction of the total channel flow noise in coolant channels containing SVEA fuel than in channels containing 8x8 lattice fuel. This difference might be expected to appear at the core resonance frequency in coherence patterns between channel flow signals and neutron flux signals. At this point, attention may be turned to figs 12 a - e, referring to the noise recording that was made while the plant power control system was operating.

A comparison between figs 12 c and 11 c shows that plant operation in the power control mode tends to enhance the peak at the core resonance frequency (0.75 Hz) in the noise auto spectrum of the APRM signal. Apparently, the actions of this control system introduce coolant flow disturbances over a wide frequency band, to which the global neutron flux responds in relation to the gain of the transfer function involved - reaching its peak value at the core resonance frequency.

It is interesting to see how the noise patterns of the channel flow signals are influenced by the enhanced neutron flux noise at this frequency. As regards the channel flow through the 8x8 lattice assembly, the noise auto spectrum of the flow signal in fig. 12 a does not differ significantly from that in fig. 11 a. The same applies to the coherence patterns presented in figs 12 b and 11 b, involving the same flow signal. However, the situation is different for the channel flow through the SVEA assembly. Here, the increased neutron flux noise at the core resonance frequency (fig. 12 c) is accompanied by a small - but noticeable - peak in the noise auto spectrum of the channel flow in fig. 12 e, which is not apparent in fig. 11 e. Concurrently, the coherence seen in fig. 12 d between the same channel flow and the neutron flux at the core resonance frequency is higher than in fig. 11 d. Apparently, the spectral peak at 0.75 Hz in fig. 12 e is neutron flux induced.

Coherence studies of neutron flux and channel flow signals have been made in several reactors, covering a variety of operating situations and involving both SVEA-64 and SVEA-100 assemblies. It is a general experience from these studies that a neutron flux induced component can be perceived in the noise patterns of the channel flows through SVEA assemblies, but not in the flows through 8x8 lattice assemblies. Apparently, the neutron flux induced flow noise in the channel flows through the SVEA assemblies occupies a larger fraction of the total flow noise than in the flows through the 8x8 lattice assemblies. This supports the conclusions made above - from the comparisons between the noise auto spectra of the various channel flow signals - that the thermal-hydraulic flow noise is higher in the channel flows through 8x8 lattice assemblies than through SVEA fuel assemblies.

7. DISCUSSION

The results presented here illustrate that useful operational experience concerning the thermal-hydraulic stability of channel flows through individual fuel assemblies in a reactor core can be extracted from the noise patterns of the channel flow signals. Moreover, information on coupling effects between channel flows and neutron fluxes can be obtained via coherence analysis of signal pairs involving flow signals and neutron flux signals. These techniques have been used to demonstrate that channel flows through SVEA demonstration assemblies are even more stable than flows through 8x8 lattice assemblies. So far, no significant differences have been noticed between the stability characteristics of channel flows through SVEA-64 and SVEA-100 assemblies.

These results have all been obtained from flow noise studies of SVEA demonstration assemblies. By now, the reactor cores concerned have received whole reloads of SVEA fuel. The flow noise studies continue, with the intention to probe what long-term consequences this may have on core stability.

REFERENCES

POWER OSCILLATIONS AT THE LEIBSTADT NUCLEAR PLANT

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Abstract - Data from the Leibstadt Nuclear Power Station at conditions near rated power and core flow, and along the rated rod line, have been used to analyze an observed fluctuation in the flow rate of each recirculation loop and the corresponding change in core flow and power. This fluctuation is characterized by alternate formation and dissipation of a vortex at the recirculation header cross.

1. INTRODUCTION

The Leibstadt Nuclear Plant is a Boiling Water Reactor (BWR-6) with an electrical output (gross) of 1030 MW. A simple overview of the reactor steam and water flow paths is shown in Fig. 1. The relationship between the operational parameters is shown schematically in Fig. 2.

In Figure 3 the piping, pump and valve layout for the recirculation system is shown in some detail.

During the startup and full power testing, a fluctuation in the electrical power output was observed. Further investigation showed corresponding changes in the reactor steam flow, core power, core flow and recirculation drive flow measurements, thus confirming that the observed power changes are not due to instrumentation malfunctions.

The forcing function for these fluctuations is a random change in the recirculation loop jet pump flows where Loop A exhibits about 2 1/2% to 3% maximum change and Loop B shows about 3% to 3 1/2% maximum change. The changes in the other major plant variables (core flow, core power, electrical power) all exhibit behavior consistent with these flow changes.

The observed flow changes occur at random intervals, but their magnitude and frequency appear to be a function of whether the recirculation flow control valve is in the automatic neutron flux control mode (with or without the flux estimator in use) or in the manual mode with no flux or drive flow feedback. Data indicate that the frequency of change is much less when the valve control system is in the manual position. Here the frequency is approximately 10 cycles per hour versus 15-180 cycles per hour during automatic flux control. However, the automatic flux mode, the normal mode of operation, will decrease the magnitude of the electrical power swings slightly and try to maintain a constant thermal power as the flow control valves respond to the flow changes.

2. SUMMARY AND CONCLUSIONS

The fluctuation in the recirculation drive flows at the Leibstadt Nuclear Power Station has been characterized by amplitude and frequency information obtained from the plant transient data acquisition system. From samples of this data it was determined that:

The period between the start of events for the recirculation drive flow varies from 20 seconds to four minutes (frequency of 180 to 15 cycles per hour) when the recirculation control system is in the auto flux mode. In manual mode this frequency decreases to about 10 cycles per hour.

The maximum amplitude of the individual jet pump flow changes is 3% to 3 1/2% of the nominal flow rate. Core flow changes can reach a maximum of 2 1/2% if both loops change simultaneously in the same direction.
Fig. 1  Steam and Recirculation Water Flow Paths
Fig. 2  Interdependence Between Operational Parameters

Fig. 3  Piping Layout for the Recirculation System
The corresponding change in reactor thermal power and generator electrical power is nominally 1% (with peaks of 1 1/2%) at a frequency as high as 180 cycles per hour in the auto flux mode. The frequency of the changes decreases to about 10 cycles per hour when the control system is in manual position mode. The auto flux mode will decrease the magnitude of the electrical swings slightly as the flow control valve (FCV) responds to maintain a constant power output.

The A loop is in the increased flow condition about 50% - 90% of the time.

The B loop is in the increased flow condition 10% - 50% of the time.

The recirculation system also experiences pressure oscillations with amplitudes of up to 0.7 bar in the frequency range of 125Hz to 370Hz. The amplitude of these pulsations seems to be influenced by the flow fluctuations and the frequencies correspond with the recirculation pump vane passing frequency (1480 RPM with 5 impeller vanes). There are differences in the pressure oscillations between Loops A and B. On the pressure side of the pump both show a frequency of 125Hz and similar amplitudes. On the suction side the A loop has a dominant frequency of ca. 370Hz where as B has 125Hz. These differences may be accounted for by the different piping design and may in turn affect the frequency, amplitude and abruptness of the flow fluctuations.

3. DESCRIPTION OF THE PHENOMENA

3.1 Introduction

In order to quantify the effects of the various plant parameters on power oscillations numerous measurements were performed.

Data was obtained from the following plant systems:

a) The process computer
b) The transient analysis system (GETARS)*
c) Various instruments in the control room

The process computer was used to define the general plant condition.

The GETARS* system was developed for the recording of relatively fast transients with a time resolution of approximately 10ms. Most measurements were made on this system.

The instrumentation in the control room was also used to define the general plant condition. In addition by observing these instruments various interdependencies could also be observed, which could later be better defined by the use of specific measurement equipment.

In addition strain gages were attached to the suction and discharge piping of the recirculation pumps to measure the pressure and look for any correlations with the power fluctuation phenomena. Finally the vibration monitoring system of the recirculation pump motors was used to identify significant vibrations and their possible correlation with the pressure or flow fluctuations.

* GETARS is a tradename of the General Electric Company.

3.2 Observations

It is observed that the jet pump flow in one recirculation loop suddenly reduces by two to four percent in less than a second. Because of the resultant reduced backpressure in the lower plenum of the Reactor Vessel the flow in the other loop generally shortly afterwards increases by approximately half to one percent. The combined effect on the core flow is then seen in a reduction of one to two percent within 1 1/2 to 2 seconds. Because in the boiling water reactor the thermal power is directly proportional to the recirculation flow the steam production is simultaneously reduced by the same amount.

The turbine control valves control the reactor pressure to a previously set value. Because of the reduced steam production this leads to a corresponding throttling of the turbine control valves. Therefore the turbine generator power is reduced approximately 5 to 6 seconds later to a new power level.

The abrupt changes in the flow in the recirculation system can occur in both loops and in both directions randomly. The effect of changes in recirculation flow and jet flow on the various other plant parameters is shown schematically in Fig. 2.
A good overview of the plant behavior during power oscillations is obtained from time history plots. Examples of such time history plots for various values of core flow are given in Figs. 4 to 7. This data was obtained from the GETARS computer system. The following parameters are shown:

- Core Flow %
- Electrical Power MW
- Recirculation Flow Loop A Kg/s
- Recirculation Flow Loop B Kg/s
- Jet Flow Loop A %
- Jet Flow Loop B %
- Valve Position FCV A %
- Valve Position FCV B %
- APRM Flux %
- Feedwater Flow Kg/s
- Steam Flow Kg/s

A closer examination was made of the jetpump flow distribution in both loops and in both high and low flow modes. The results are summarised in Figs. 8 and 9. This shows that the low flow mode is characterised by a higher flow in the central two jet pumps (3/6 or 5/16) and low flow in the remainder. In the high flow mode the flow is more evenly distributed.

In Fig. 10 and 11 the jet pump flow is plotted as a function of flow control valve position. Here can clearly be seen the bistable character of the jetpump flow of the recirculation loops A and B. For a given control valve position there are a range of possible flow levels which are however only stable close to the upper or lower bounding curves. We have not attempted in Figs. 10 and 11 to describe the measured values by best fit curves because the coreflow does not increase uniformly with valve position as would be normally expected. There are regions where the jet pump flow is lower than expected. For example in Loop A this is clearly seen between the valve position 50% - to 65%. In Loop B the same effect occurs between 76% and 88%.

The amplitude of the jetpump flow fluctuations as a function of total core flow is shown in Figs. 12 and 13. A marked increase above 80% core flow is apparent, with a large scatter in the observed values. At certain core flows "quiet" periods with low amplitudes are observed. The heavy lines in Figs. 12 and 13 show the contribution from instrument noise and flow control system effects.

3.3 Frequency Analysis

Fig. 14 to 15 show the results of the strain gauge measurements on the recirculation suction and discharge lines in the time domain.

The results of a FFT Analysis of these signals are shown in Table 1 and in Figs. 16 and 17.

<table>
<thead>
<tr>
<th>Frequency [Hz]</th>
<th>Location</th>
</tr>
</thead>
<tbody>
<tr>
<td>25</td>
<td>50</td>
</tr>
<tr>
<td>Loop A Suction</td>
<td>0.08</td>
</tr>
<tr>
<td>Discharge</td>
<td>0.16</td>
</tr>
<tr>
<td>Loop B Suction</td>
<td>--</td>
</tr>
<tr>
<td>Discharge</td>
<td>0.08</td>
</tr>
</tbody>
</table>

These oscillations are permanently present with slightly varying amplitude. There are some differences between Loop A and B. Loop B shows in-phase oscillation of the suction and discharge lines at 125 Hz whereas Loop A shows 370 Hz in the suction and 125 Hz in the discharge line.

The strong coupling of a 125 Hz wave in Loop B may be caused by an interconnection from another piping system. The length of the connecting line may be such as to set up a standing wave of this frequency.

PNC- AA**
Fig. 4  Sample of Data Showing Flow Anomaly - Loop A (Auto Flux Mode) (96% Core Flow)

Fig. 5  Sample of Data Showing Flow Anomaly - Loop B (Auto Flux Mode) (96% Core Flow)

Fig. 6  Sample of Data Showing High Frequency Flow Changes (Auto Flux Mode) (93% Core Flow)

Fig. 7  Sample of Data Showing Flow Anomaly - Loop B (Manual Mode) (99% Core Flow)
Fig. 8 Recirc Loop A  Jetpump Flow Distribution

Fig. 9 Recirc Loop B  Jetpump Flow Distribution
Fig. 10  Flow Control Valve Position [%]

Fig. 11  Flow Control Valve Position [%]
Fig. 12  Delta Jetflow Loop A

Fig. 13  Delta Jetflow Loop B
Fig. 14 Strain Gauge Measurements from Recirculation Loop A

Fig. 15 Strain Gauge Measurements from Recirculation Loop B
Fig. 16 Frequency Analysis of Strain Gauge Measurements from Loop A

Fig. 17 Frequency Analysis of Strain Gauge Measurements from Loop B
The pressure oscillations seem to be symptoms of different velocity-distributions in the pump discharge lines and might explain the different behaviour of the flow fluctuations in the riser lines.

The information hidden in the lower frequency region (≤70 Hz) has not yet been fully analysed. It may be of a more stochastic nature produced by fluid oscillations.

This frequency analysis may help to explain and understand the hydraulic phenomena occurring in the recirculation loops. Further measurements are planned to help correlate the flow fluctuations and the pressure oscillations.

4. CAUSES OF THE FLOW FLUCTUATIONS

As mentioned at the beginning of the report the cause of the observed fluctuations in flow and power is the occurrence of two stable flow conditions in the external recirculation loops.

It is believed that a stable vortex forms in the reducer just above the header cross (Fig. 3) so that the flow to jet pumps 5 and 6 (or jet pumps 13 and 16) is reduced. At the same time the flow to the other four risers is correspondingly increased so that the total jet pump flow for this recirculation loop will be at the maximum. This is the so-called high flow mode. The low flow mode is produced when this vortex moves downwards and into the header where it forms a helical vortex which propagates along the header and reduces the flow to the four exterior risers.

The stability of the various vortex configurations is dependent on several parameters.

The most important of these are:

- total core flow
- position of the control valves
- pressure drop over the core plate
- pressure changes in the recirculation system
- relationship between the recirculation flows of loop A and B

These parameters are either fixed or can only be changed within limited ranges. It is also not possible to predict the influence of changes in these parameters on the power oscillations.

5. ACKNOWLEDGEMENTS

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REACTOR NOISE STANDARDS IN THE U.S.A.

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Abstract - The American Society of Mechanical Engineers (ASME) has fostered the development of codes and standards to be used by all facets of the general industry. The ASME Committee on Operation and Maintenance of Nuclear Power Plant (O&M Committee) currently has responsibility of 20 ASME nuclear power standards addressing areas such as "Requirements for Inservice Performance Testing of Pressure Relief Devices" (OM-01) to "Inservice Testing of Diesel Drives in Nuclear Power Stations" (OM-16). The Subcommittee on Vibration Monitoring (SCVM) is responsible for several O&M Committee standards, two of which utilize reactor noise techniques: OM-05, "Inservice Monitoring of Core Support Axial Preload in Pressurized Water Reactors", and OM-12, "Vibration Monitoring and Diagnostics of Light Water Reactor Loose Parts". This paper presents an outline of the O&M Committee standards and briefly describes the highlights of the two SCVM standards which utilize Reactor Noise techniques.

1. INTRODUCTION

With the advent of commercial nuclear power, the American Society of Mechanical Engineers (ASME) established the Board on Nuclear Standards. The Board on Nuclear Standards has a rather comprehensive scope, in particular the following areas:

"To manage all ASME activities related to codes, standards, and accreditation programs directly applicable to nuclear facilities and technology...To develop recommended ASME positions with interfacing bodies, including regulatory agencies, on matters related to nuclear facilities and technology." (ASME AS-11)

The Board on Nuclear Standards recognized the importance of operation and maintenance activities at nuclear power plants and established the Committee on Operation and Maintenance of Nuclear Power Plant Components (O&M Committee). The O&M Committee scope of activity is:

"To develop, maintain, and review codes and standards that are considered necessary for the safe and efficient operation and maintenance of nuclear power plants, particularly as they relate to structural and functional adequacy." (ASME AS-11)

Currently the O&M Committee has approved the scope for 20 standards whose titles are listed in Table 1. Of these scopes, the first five were fully developed and issued as ASME standards. Furthermore, these five were also adopted by the American National Standards Institute, Inc. (ANSI).
Table 1. Titles of Nuclear Standard Scopes approved by the ASME Operations and Maintenance Committee

| OM-01 | Requirements for Inservice Performance Testing of Pressure Relief Devices |
| OM-02 | Requirements for Performance Testing of Nuclear Power Plant Closed Cooling Water System |
| OM-03 | Requirement for Preoperational and Initial Start-Up Testing of Nuclear Power Plant Piping Systems |
| OM-04 | Examination and Performance Testing of Nuclear Power Plant Dynamic Restrains (Snubbers) |
| OM-05 | Inservice Monitoring of Core Support Axial Preload in Pressurized Water Reactors |
| OM-06 | Performance Testing of Pumps |
| OM-07 | Requirement for Preoperational and Initial Start-Up Thermal Expansion Testing of Nuclear Power Plant Piping Systems |
| OM-08 | Periodic and Performance Testing of Motor-Operated Valves (MOV’s) |
| OM-09 | Inservice and Performance Testing of Cranes |
| OM-10 | Performance Testing of Valves |
| OM-11 | Vibration Monitoring of Heat Exchangers |
| OM-12 | Vibration Monitoring and Diagnostics of Light Water Reactor Loose Parts |
| OM-13 | Requirements for Periodic Performance Testing and Monitoring of Power Operated Relief Valve Assemblies (PORV’s) |
| OM-14 | Vibration Monitoring of Rotating Equipment |
| OM-15 | Requirements for Performance Testing of Nuclear Power Plant Emergency Core Cooling Systems for PWR’s |
| OM-16 | Inservice Testing of Diesel Drives in Nuclear Power Stations |
| OM-17 | Requirements for Performance Testing of Nuclear Power Plant Instrument Air Systems |
| OM-18 | Startup and Periodic Performance Testing of Electro-Hydraulic Operated Valve Assemblies Used in Nuclear Power Plants |
| OM-19 | Startup and Periodic Performance Testing of Electro-Pneumatic Operated Valve Assemblies Used in Nuclear Power Plants |
| OM-20 | Requirements for Performance Testing of Nuclear Power Plant Emergency Core Cooling Systems for BWR’s |

In 1976 the O&M Committee formed the Subcommittee on Vibration Monitoring (SCVM) to address standards which required use of techniques and methods to measure and monitor non-stationary structural-mechanical properties. Of the standards listed in table 1, the SCVM has responsibility for OM-03, OM-05, OM-07, OM-11, OM-12, and OM-14.
2. REACTOR CORE SUPPORT AXIAL PRELOAD MONITORING STANDARD

ASME Standard OM-5 "Inservice Monitoring of Core Support Axial Preload in Pressurized Water Reactors" is based on the technique of interpreting the signal fluctuations of ex-core ion chambers caused by core reactivity changes and neutron attenuation due to lateral core motion. This technique is more commonly referred to as "Neutron Noise". This technique is one of the original applications of noise analysis and relies on classical noise analysis techniques involving the use of the normalized power spectral density (NPSD) and normalized cross-power spectral density (CPSD) functions as well as wide and narrow band normalized root mean square (NRMS). A discussion of neutron noise analysis is beyond the scope of this paper and the interested reader is referred to a number of excellent technical papers on the subject published in the transactions of SMORN meetings. It should be noted that the standard includes tutorial appendixes on the theoretical basis of neutron noise as well as on data acquisition and reduction techniques.

The OM-5 standard is intended to detect significant loss of axial preload of a typical light water pressurized water reactor where the core support assembly is suspended from a ledge situated on the upper part of the reactor vessel and restrained by the reactor vessel head assembly. The standard suggests that a loss of axial preload could occur due to abnormal wear at the reactor vessel to core barrel mating surface (a long-term change) or improper installation of the reactor internals (a short-term change). The program described in the standard is divided into three basic phases: a baseline phase, used to establish reference data; a surveillance phase, used to detect deviations from acceptable values; and a diagnostic phase, were deviations detected during the surveillance phase are investigated.

The baseline reference data is recommended to be acquired at the beginning, middle, and end of each of the first three fuel cycles of a new plant or during the same times of the first fuel cycle that the monitoring program is implemented. In addition, the standard points out that a new set of baseline reference data is required when significant changes are made to the core, reactor internals, operating conditions, and after every core barrel removal. The type of data to be acquired is specified as the time history and DC level of all functioning ex-core power range detectors, single section or summed signal form upper and lower sections. Acquired data must be able to permit obtaining the the cross-power functions (NCPD, coherence, and phase) for all cross-core and adjacent pairs of detectors as well as the normalized power spectral density for each detector. The standard suggest a minimum frequency band of 0.2 - 20 Hz with a frequency resolution of at least 1 percent of the highest calculated frequency of interest. The baseline reference data is used to establish the beam mode center frequency as well as two frequency ranges: a narrow band encompassing +/- 25% of the beam center frequency, and a wide band extending from 0.2 to 20 Hz. The establishment of these two frequency ranges are for use in the monitoring phase. The standard recommends that the center frequency of the core barrel motion should be verified by analysis (structural-dynamic calculations), data acquired during a preoperational reactor internals vibration measurement program, or by comparison with data from a similarly designed and constructed reactor whose core barrel motions have been verified analytically or by measurements.

The surveillance phase periodically confirms that the neutron noise NRMS values and beam mode center frequency are within predetermined limits established during the baseline phase. The data is required to be acquired at the start and end of each fuel cycle and at every 90 effective full power days (EFPD) during the cycle. The standard permits the use of either a continuous surveillance monitoring system or the acquisition of a set of data of the same type delineated for the baseline phase. The determination of a potential anomalous condition is based on comparisons of the new narrow and wide band NRMS values with those established during the baseline phase. Guidelines for establishing criteria for further analysis are provided in an appendix to the standard with the acceptable limits established by the plant owner.

Should the neutron noise NRMS values deviate from pre-established limits, the standard delineates the requirements of a diagnostic phase to determine if the deviations are due to changes in core barrel motion (indicative of loss of axial restraint) and to establish further actions. The data required as well as the analysis are the same as those performed in the baseline phase. The diagnostic data
set are compared to the baseline and surveillance set with the aim of detecting deviations and trends. Two appendixes are provided to assist in the data evaluation and evaluation of baseline signal deviations. One of the appendixes provides illustrations of typical signatures, baseline NPSD ranges, as well as an example of an auto-power spectral density of a plant with loss of pre-load. The standard suggests that the results of this phase shall indicate whether or not the intervals of acquiring data should be shortened or whether changes to the monitoring methods should be implemented.

3. LOOSE PARTS MONITORING STANDARD

ASME Standard OM-12 "Vibration Monitoring and Diagnostic of Light Water Reactor Loose Parts", also known as "Light Water Reactor Loose-Part Monitoring and Diagnostics" is presently undergoing internal committee review and balloting. This standard has as its genesis the need to provide guidance in implementing and supplementing the United States Nuclear Regulatory Commission Regulatory Guide 1.133 "Loose-Part Detection Program for the Primary System of Light-Water-Cooled Reactors" issued in May 1981. The standard is based on the reactor noise techniques of detecting and analyzing structural stress waves (or noises) commonly referred to as "Loose Parts Monitoring". As in the case of the neutron noise standard, discussion of loose-parts techniques is beyond the scope of this paper and the reader is once more referred to SMORN transactions.

The standard describes some of the problems of presently installed systems as part of the introductory section and points out that high false alarm rates have been a major generic problem for loose parts monitors in the United States. These high false alarm rates have caused a reduced confidence in the information obtained from loose-parts monitors. The standard addresses this issue by recommending that the system sensitivity be set on the basis of the installed system background noise to achieve the maximum sensitivity commensurate with an acceptable false alarm rate. This approach in setting system sensitivity is in contrast to previously accepted methods involving fixed sensitivity levels based on correlations with impact energies.

The standard discusses the techniques available for loose parts monitoring and diagnostics and provides practices for optimizing installed system performance, for establishing and maintaining calibration / alert levels, and for performing baseline and periodic data acquisition and reduction. A brief tutorial in the body of the standards addresses the characteristics of metallic impact stress waves and provides insight of the interrelationship of acceleration spectral content, impact velocity, loose-part mass, and impact duration. The section on equipment addresses a rather comprehensive list of topics ranging from sensor types and mounting methods, sensor arrays, to signal conditioning and analysis equipment requirements. The recommended sensor array for a pressurized water reactor consists of: 3 accelerometers located approximately 120° apart around the top of the reactor vessel at an elevation between the vessel core barrel support flange and the reactor vessel head lifting lugs; 3 accelerometers located approximately 120° apart around the lower section of the reactor vessel and mounted on in-core guide tubes within 18 inches of the reactor vessel and 2/3 the radial distance outward from the reactor vessel axis (on plants without lower incore guide tubes the sensors shall be mounted directly to the reactor vessel); and 3 accelerometers mounted on each steam generator (2 in the vicinity of the tube sheet). For boiling water reactors, the recommended sensor array consists of 12 accelerometers, three each at the main steam outlet elevation, feedwater inlet elevation, recirculation suction elevation, an reactor vessel bottom.
The standard section addressing program elements suggests methods to reduce personnel radiation exposure, equipment calibration requirements, baseline impact testing, initial monitor set-points, periodic monitoring and testing, and alarm response and diagnostic. The baseline impact testing, performed during cold-shutdown plant conditions, discusses two acceptable methods: pendulum/ball drop and instrumented hammer. These techniques are discussed in one appendix. The impact locations suggested are at least two test locations at each natural loose-parts collection region and the secondary side of each steam generator. The baseline data analysis requires the frequency response function for force hammers displaying the ratio of acceleration (response) to force (input) or auto-spectra of each sensor response to a pendulum/ball drop. In addition, the delay times between wave arrival at different sensor locations should be measured for all channels.

The initial monitor alert level set-points is recommended by the standard to be set at either 3 times the long term, band-limited background noise; 1 g; or the manufacturer's recommendation. Individual channel alert set-points are to be adjusted for minimizing false alarms after reaching power operation. The recommended false alarm rate is suggested not to exceed 1 false alarm indication for the system approximately every 2 weeks. The standard cautions that set-points for minimizing false alarms be checked so as not to compromise sensitivity to potentially damaging loose parts.

Periodic monitoring requirements of the standard is delineated for every operations daily shift, once a week, once a quarter, and at the end of each fuel cycle. The once per shift monitoring entails essentially listening to the audio output of each channel for anomalous noises. Each week the standard recommends measuring and documenting the background level of each channel and performance of a monitor system self-check. Each quarter, data should be recorded for each channel for historical purposes, acquisition of power spectral densities for each channel and comparison with data from the two preceding quarters, and measuring and documenting the voltage and current to each remote charge amplifier (if installed). At each refueling outage, the standard requires that an evaluation be made for component degradation and corrective action implemented.

The section of the standard on alarm response and diagnostic outlines actions that should be taken to determine the cause of the alarm. The standard suggests, in addition to listening to all monitor channels, attempting to reset the monitor and verify that a data set is being recorded. Should there be indications of metallic impacting, the standard suggests that appropriate personnel perform a diagnostics evaluation to estimate the location of the metallic impact, estimate impact energy and mass, and estimate the damage potential and the probable result of continued plant operation.

4. OBSERVATIONS

The advent of nuclear reactor standards based on reactor noise techniques reflects the maturation of the reactor noise technical specialty. OM-5, the axial core barrel pre-load monitoring standard, is a direct application of a technique which was once was strictly experimental and in the realm of specialists. OM-12, the loose-parts monitoring standard, reflects considerable operational experience with rather primitive monitors and provides much needed guidance on the implementation of a functional loose-parts monitoring program. Both standards owe their methodology to contributions of countless specialists from a variety of countries. Other standards using reactor noise techniques are already being discussed informally based on preliminary results of a few investigations. Plant life extension criteria by noise analysis could become a reality in the foreseeable future.
NEW TRENDS IN VIBRATION AND ACOUSTIC MONITORING IN NUCLEAR COMPONENTS IN EDF

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Abstract - Main points of feedback experience obtained by EDF since 1984, in the area of 900 MW and 1300 MW units pressurised water reactors surveillance, are firstly commented. The monitoring programs objective evolution is presented from projects, concerning sensors maintenance conditions upgrading and new monitoring systems development. Current R and D programs are then presented. They concern new digital techniques application for plant operator assistance and for automatic monitoring. Some future R and D programs are lastly proposed to improve reactors critical components surveillance.

Key words - Reactor, surveillance, evolution, sensors, maintenance, monitoring systems, new digital techniques, operator assistance, critical components.

INTRODUCTION

Programs initiated by EDF in the field of pressurised water reactors surveillance concern reactor internals mechanical behaviour monitoring, loose parts detection in primary cooling system and sensors monitoring. Main features of EDF experience feedback collected in this area since 1984 are firstly commented. Then, current monitoring projects objective evolution is shown. R and D programs in progress or proposed for the near years are lastly presented.

1. EDF EXPERIENCE IN THE FIELD OF PRESSURISED WATER REACTORS SURVEILLANCE

This experience is large; reactors monitoring procedures and systems have been defined and installed since the starting-up of the first EDF PWR units in 1977, and monitoring data have been collected, processed and analysed for about 10 years.

In this chapter, methods applied and systems used are recalled and main features occurred in that field are commented. They concern in particular the experience of reactor monitoring systems operation by nuclear plant people and the presentation of complementary knowledge collected for 1300 MW and 900 MW reactors surveillance.

1.1. Systems

Since 1977, all operating EDF PWR units (thirty three 900 MW units and nine 1300 MW units, on July 1, 1987) are equipped with specific monitoring systems supplying reactor internals mechanical behaviour monitoring and loose parts detection, and periodic or on-line monitoring procedures are also applied.

Reactor internals mechanical behaviour surveillance is performed by the SURVIB system, part of the KIR system installed on all 900 MW PWR units. Monthly, quarterly and annual monitoring procedures have been defined with plant people and are applied. Noise analysis technique are applied. Sensors used are accelerometers mounted at the bottom of reactor vessel and ex-core neutron detectors. An upgraded model of that system, able to perform a continuous monitoring have been designed and is installed on 1300 MW units. A complementary surveillance is periodically performed on some reactors by means of in-core neutron detection noise analysis. This monitoring concern mechanical behaviour of critical components (fuel assemblies, core instrumentation thimbles) and anomalies detection (baffle jetting).

Loose parts detection is performed on 900 MW units by two subsystems: the ACCOUSUR subsystem, which allows a periodic acoustic monitoring, and the DEVIANIY system, which allows a continuous acoustic monitoring. Acoustic signals are given by accelerometers located in primary and secondary coolant systems sensitive areas.
The DEVIAN T device is able to detect every acoustic signal; the level of which is higher than a reference value and to generate an alarm in the control room. On 1300 MW PWR units, a new model of this device has been developed; it is described in a following chapter.

For sensors monitoring, first projects achieved concerned resistance thermometers and pressure transducers. Current projects concern neutron sensors.

1.2. Reactor monitoring systems operation experience

During 1986, an enquiry has been held among EDF nuclear plants about reactor monitoring systems operation experience. This experience is large: more than 200 reactors-operation years. In 1987, results of the enquiry have been analysed (1). Main conclusions are:

- good acceptance of these systems by nuclear plant operators; operation and maintenance are easy.
- necessity of a permanent, even intensive training for plant people involved in these systems operation and in the monitoring data interpretation.
- request of a better presentation of monitoring procedures to be applied and of monitoring data to be processed and analysed.

1.3. 1300 MW reactors internals vibration monitoring

1300 MW reactors internals vibratory behaviour has been characterized.

Firstly, an investigation tests campaign has been carried out during hot tests Paluel 1 unit. These site measurements used a specific instrumentation: accelerometers and strain gauges temporarily installed on reactor internals (core barrel, ...) to allow to compare the experimental behaviour obtained during site tests and the theoretical behaviour calculated by mathematic model during design. During these tests, information given by permanent instrumentation (external accelerometers mounted on vessel) is also calibrated from signals emitted by temporary instrumentation.

Secondly, during Paluel, Flamanville, Saint-Alban and Cattenom plants start-up, when units have reached nominal load, on site measurements have been performed from external accelerometers and from ex-core neutron sensors. These measurements gave reactor components vibratory signature reference.

Besides, periodical measurements have been performed on all operating reactors.

1300 MW reactors internals vibratory behaviour is actually characterized by two distinct situations:

- either a permanent contact of core barrel on radial keys,
- either a free swinging of core barrel (mode 1 at 8.5 Hz; mode 2 at 20 Hz) with intermittent contact on radial keys.

1.4. 1300 MW and 900 MW reactors thimbles vibration analysis and monitoring

In March 1985, during a Paluel 1 reactor periodic flux map test, a leak occurred on two thimbles. Then, a thimble vibration analysis and monitoring program has been initiated (2).

For thimble vibration analysis, a first method has been developed and applied on 1300 MW units; this method is based on the analysis of shocks detected by accelerometers temporarily installed on thimbles tubes in-core instrumentation room and connected to KIR system. Twenty measurements campaigns have been performed in 1985 and 1986 on nine reactors in order to evaluate modifications proposed to reduce thimbles vibratory excitation.

A second method has been experimented on Paluel 1 and 2 reactors. It is based on spectral and time analysis of in-core neutron chambers noise. An estimation of the thimble modal shape in the fuel assembly instrumentation tube has been made.

900 MW reactors are also concerned by thimbles vibrations phenomena. However, they are less sensitive to this phenomena than the first 1300 MW units, so a monitoring program has been initiated.

During 1986, thimbles vibratory measurements have been performed on 9 reactors (7 in France, 2 in Belgium) from accelerometers temporarily installed.

Since August 1986, Chinon B3 unit has been equipped with accelerometers mounted on 50 thimbles guide tubes, and an on line monitoring system, SATIN, has been installed on site. This system allows
to know kinetics of guide tube wear associated with thimbles shocks and to determine a correlation between the number of detected shocks and the wear estimated by Eddy current test during annual refuelling.

In 1987, on this unit and on some other units (Tricastin 4, ...) in-core neutron detector noise analysis is also experimented to characterize thimble vibratory behaviour.

1.5. Reactor fuel assemblies vibration monitoring

In-core neutron detector noise analysis allows to characterize different fuel assemblies vibratory behaviour.

Several EDF 900 MW reactors (Bugey, Blayais, Dampierre, Tricastin, ...) have been refuelled with Exxon, KWU or AFA fuel assemblies different from the original Frasema fuel assemblies.

The fuel assemblies vibratory signature has been collected. So, at Dampierre Nuclear Plant, fuel assemblies vibratory behaviour is characterized by a mode 2 frequency 7 hz for KWU fuel assemblies and by a mode 2 frequency 6 hz for Frasema fuel assemblies.

This surveillance may be also performed from ex-core neutron detector noise analysis. This last method is applied at Dampierre Nuclear Station, where an automatic system, SACRE, allows to collect reactor fuel assemblies vibratory data 4 times a day.

1.6. Loose parts detection

Since 1984, the DEVIANT loose parts monitoring and detection system allowed the detection of two loose parts in two nuclear plants (4).

In September 1985, at Fessenheim 1 unit, a loose part was detected ; diagnostic of loose part location in steam generator number 1 water box was done and loose part mass was estimated at about 80 grammes. Unit operation continuation was decided until 1986 annual refuelling, and an intensive monitoring program was initiated. In August 1986, the steam generator water box was inspected. It was found a 75 grammes loose part, piece of an electric torch holder used during 1985 annual refuelling. No damage was visible.

In March 1987, at Tricastin unit 4, during reactor starting up after a unit shut down for a repair on a reactor auxiliary system, a loose part was detected in the water box of a steam generator. The mass of the loose part was evaluated larger than 80 grammes, after analysis of recorded data by specialists. Unit was then shut down. The steam generator water box was inspected and a screw of a control rod guide tube was found. This piece of 125 grammes mass did not cause any damage.

These two incidents demonstrate the usefulness of loose parts monitoring systems for incipient failure detection.

2. EVOLUTION OF REACTORS MONITORING PROGRAMS

First monitoring programs initiated had for main objective anomaly detection. Actually, plant operators are particularly interested by the development of new methods, which may provide information for components maintenance upgrading, and by the development of new monitoring system, which may provide a machine aided surveillance data interpretation and diagnostic, when an anomaly occurs.

Thus, two programs under development in EDF illustrate the aim ; the first concerns neutron sensors monitoring, the second is relative to the development of a new 1300 MW PWR unit loose parts monitoring and detection system.

2.1. Neutron sensors monitoring

Several monitoring methods are experimented with the aim of predictive maintenance.

Firstly, periodic verification of cables and connections conditions is performed by insulation test or by reflectrometry. This last method allows to detect and localise failures (break, short circuit, ...) and some degradations (bad connections, ...).

Secondly, three neutron sensors monitoring programs have been undertaken (3) :

- a passive method, based on neutron noise analysis is experimented on several plants (Bugey, Fessenheim) ; a sensor functioning descriptor is determined by signal spectrum analysis techniques in a large frequency range (several Khz) ; a descriptor variation is representative of a sensor performances degradation (aging).

- another method is based on the interpretation of the ascending part of the saturation curve of
an ionization chamber. If aging occurs, the saturation curve is obtained for a higher voltage: This method is applied for the verification of intermediate and powers range detector.

- a third method is used for the verification of normal source detector. It is based on the shape of the discrimination curve at nominal voltage of this chamber; the slope of the curve is determined; any increasing of this slope is representative of a chamber degradation.

2.2. 1300 MW pressurised water reactors loose parts monitoring and detection system

A new loose parts monitoring and detection system, IDEAL, using microelectronics and digital techniques, has been developed and installed on EDF 1300 MW PWR units (4).

The performances of this system are higher than those of the DEVIAN'T system, which equips EDF 900 MW PWR units.

Indeed, in the DEVIAN'T system, three main functions are performed:

- continuous monitoring of reactor cooling system;
- detection of abnormal acoustic noise; anomaly detection criterion is a peak amplitude level;
- triggering of an alarm in control room, when an anomaly occurs.

In addition to these functions, IDEAL system provides the following supplementary functions:

- location of the area affected by the loose part;
- storage of data collected by the system for 4 weeks;
- prediagnostics in case of anomaly; the system processes and records data relative to the time of anomaly occurrence and to the factors characterizing the anomaly (amplitude, duration, impulse frequency content, time intervals between impacts);
- perturbography. When a defect occurs, data concerning the sensors of the affected area are automatically recorded and stored on a magnetic tape, and may be analysed on a display console;
- autoverification of the system. Periodically, plant operators may verify system functions by means of a specific simulation module.

During the system design, man-machine interface conditions, using display console keyboard, have been particularly investigated and analysed.

3. CURRENT DEVELOPMENTS

These programs concern application of digital techniques to aid plant operators during reactor monitoring tasks to be performed. Two programs are presented.

3.1. Expert system MIGRE aid for loose parts detection and mechanical impulse interpretation (4)

This application has for objective operator aid for signals interpretation, when the KIR system detects an anomaly from the signals given by accelerometers mounted on the bottom of reactor vessel or steam generators.

Signals interpretation and processing for diagnostic need specialists knowledge. Artificial intelligence techniques are used for the design of the system.

Expert system rules base is under creation. It comprises 150 rules. Diagnostic strategies have been defined. Tasks actually undertaken concern man-machine interface conditions definition (diagnostic software easy to be used, convenient display, ...). A prototype of the system will be tested by specialists at the end of 1987 and experimented in a Nuclear Power Plant in 1988.

3.2. PWR plant monitoring and diagnostic aid center

In order to make easier operator tasks for the monitoring of a 900 MW main equipments, the design of a PWR plant monitoring and diagnostic aid center has been undertaken by EDF.

The prototype of this center has the following functions:

- reactor coolant system loose parts monitoring and detection;
- reactor coolant pumps vibratory monitoring;
- turbine generators mechanical behaviour monitoring.

Loose parts monitoring and detection system, DEVIAN'T, is connected to the PWR plant monitoring and diagnostic aid center by means of a specific interface module.

This center will be for plant operators a specific tool, which allows:
- to automatically collect monitoring data;
- to obtain a synthetic survey of monitored equipments operating conditions;
- to store the raw sensors data and the processed data;
- to obtain diagnostic aid for the localisation and the determination of the cause of a detected anomaly, by a specific processing;
- to be connected with specialists, outside the plant, who can help units operators to interpret situation, which have not been seen before.

A prototype of this plant center will be installed in a nuclear station at the end of 1988.

4. FUTURE DEVELOPMENTS

Programs performed until now by EDF allowed to develop monitoring methods and systems for different reactor components (internals, fuel assemblies, thimbles guide tubes, ...).

For the next years, development programs concern mainly two points:

- the monitoring of mechanical behaviour of reactor control rods and control rods drive mechanism; methods experimented are respectively neutron noise analysis and acoustic techniques.

- the experimentation of different noise analysis methods applied to temperature and pressure sensors, for a better knowledge of reactor and reactor coolant system thermal hydraulics conditions.

- the surveillance of reactor coolant system components (reactor coolant pumps, primary pipes, ...) mechanical behaviour.

5. CONCLUSION

Programs performed since 1984 by EDF in the field of pressurised water reactors monitoring have mainly contributed to obtain a complementary knowledge about 900 MW and 1300 MW reactor internals vibratory surveillance.

By the cooperation between plant operators and utility monitoring specialists, the new surveillance programs objective aims at maintenance conditions upgrading and diagnostic aid.

Applications of digital techniques and artificial intelligence techniques for reactor surveillance concern mainly automatic monitoring system development and diagnostic aid to identify abnormalities.

The knowledge of critical reactor components mechanical behaviour evolution and aging has to be thoroughly studied; new analysis and monitoring programs may be proposed. International technical exchanges in this field will be of the highest interest.

REFERENCES


BENCHMARK DISCUSSION

Session chairman: H. van Dam (Netherlands)
SUMMARY OF SESSION

The present benchmark exercise is the third one in a series of three consecutive exercises in connection with SMORN meetings. At SMORN-III a so-called computational benchmark was organized for comparison of methods for the calculation of characteristic noise functions. This was followed by physical benchmarks on the occasion of SMORN-IV. These benchmarks were focused on the extraction of physical parameters from noise signals. The present benchmarks can be considered as an extension of the previous ones, in the sense that they aim at the determination of physical parameters but in addition are intended for anomaly detection.

Two benchmarks have been produced, both by groups in the Netherlands:

- artificial noise data, produced by a group from the Interfaculty Reactor Institute of Delft University of Technology,
- actual noise data from the Borssele NPP, prepared by the Energy Research Foundation at Petten.

The first benchmark was analyzed by seven groups from five different countries, whereas nine groups from seven countries participated in the second benchmark. The artificial benchmark was focused on system identification, preferably by application of MAR-methods, followed by detection of two consecutive anomalies. The main advantage of the application of computer simulation for the production of noise signals is that all relevant parameters of the system are exactly known, thus furnishing a firm basis for comparison.

The analysis of results from the participants enabled a number of conclusions to be drawn as follows. The choice of appropriate sample time and model order in AR-analysis is still an important issue. A non-optimal choice leads to pseudo-correlation between residual noise sources. It is also apparent that the strength of these sources is strongly dependent on the sampling interval. Problems were encountered with regard to normalization of signals and conversion to physical units, but the exact cause is not yet clear. The identification of physical parameters of a multiparameter system, even if this system is relatively simple, is still a problem and the accuracy of the results is in many cases rather limited. The results for the anomaly detection part are very interesting but need a more detailed assessment. The anomalies in the benchmark have a realistic character (vibrating control rod and a gradually deteriorating system stability, respectively) and they are not too conspicuous. The participants have been rather successful in anomaly detection, but it would be very useful to further analyze the characteristics of the different detection methods, in particular with regard to the inevitable compromise between false alarm rate and probability of alarm failure.

The Borssele benchmark concerns a real anomaly in the form of a rather drastic operational transient initiated by a sudden electrical load change. This change caused heavy oscillations at the generator side, propagating via the turbine into the entire secondary system and through the steam generator to the primary system. During this propagation the signal deviations due to this anomaly are gradually attenuated. The benchmark contains eight signals ranging from generator power to core coolant temperature. The anomaly in the generator power signal is very drastic and forms a well-defined time point for the start of the anomalous event. It is evident that all participants detected the event successfully and also the propagation of the disturbances was clearly identified.
There is, however, a considerable spread in the timing of the starting point as given by the participants, which might point to deficiencies in the copying of the signals or ambiguities in definitions. The total duration of the anomaly was correctly identified, which supports the conclusion that the anomaly was correctly detected; further analysis may reveal the cause of the starting time discrepancies. A wide range of methods for anomaly detection was applied by the participants like the use of moments of amplitude distributions, forms of correlation functions, power spectral densities, noise power in particular frequency intervals, pattern recognition and so on.

As a general conclusion it can be stated that the benchmark tests were useful. More benefit can however be obtained if more participants will join and the present experience is fed back to the participants for re-evaluation of their results. It is therefore the intention to proceed with these benchmarks. This will also give the opportunity to solve some problems with regard to the quality of the tapes and the incidental lack of information to participants. It is recommended that these benchmarks materials be included in the files of the Nuclear Energy Agency Data Bank as reference material for future users. This should be in the form of tapes with the original signals, but also intermediate results of analysis should be included like sampled data from the analog signals, correlations functions, AR coefficients, noise contribution ratios, transfer functions and time constants etc, obtained from fitting procedures. It should be stressed that the basic material to start from is the set of analog signals, but if a user has no sampling equipment at his disposal he may start from material of the next level, i.e. sampled data with a preferably short sampling time interval, furnished on tape.

As a final conclusion it should be stated that there is a need for new physical benchmarks after full exploitation of the present ones. These benchmark exercises should however be decoupled from SMORN meetings and should be used to create a noise signatures library at the NEA Data Bank.

Finally, thanks are due to all participants for their willingness to take part in the SMORN-V benchmark exercise and, in doing so, to contribute to the international exchange of expert knowledge in this field.
SUMMARY OF THE BENCHMARK TEST ON
ARTIFICIAL NOISE DATA

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Abstract - A survey is given of the SMORN-V artificial noise benchmark
test for checking autoregressive modelling of noise and testing anomaly
detection methods. A detailed description of the system used to generate
the signals is given.
Contributions from 7 participants have been received. Not all par-
ticipants executed both the tests on the stationary data and the anomaly
data. Comparison of plots of transfer functions, noise contribution
ratios and the spectrum of a noise source obtained from AR-analysis
partly shows satisfactory agreement (except for normalization), partly
distinct disagreement. This was also the case for the several parameters
to be determined numerically. The covariance matrices of the intrinsic
noise sources showed considerable differences.
Participants dealing with the anomaly data used very different methods
for anomaly detection. Two of them detected both anomalies present in
the signals. One participant the first anomaly and the other participant
the second anomaly only.

1. INTRODUCTION

After the "computational" benchmark test in connection with SMORN-III (Tokyo, 1981) and the
"physical" benchmark test in connection with SMORN-IV (Dijon, 1984) it was decided to
proceed with another noise benchmark test in connection with SMORN-V (München, 1987), in-
cluding a test on anomaly detection and also further tests on autoregressive modelling
of noise. Besides a test with actual reactor noise data (Türkcan, 1988), artificially generated
benchmark signals are especially suitable for such a task, as the functions or parameters to
be determined from an autoregressive model are known beforehand and anomalies in the signals
can be introduced in a controlled way to compose any degree of difficulty for detection
methods.
Therefore, a (digital) reactor core simulator was used to generate noise signals with cer-
tain characteristics. The signals are stationary over a specified period of time, after
which certain anomalies were introduced in the system by changing the characteristics of a
noise source or a transfer function.

2. OBJECTIVES OF THE ARTIFICIAL NOISE BENCHMARK TEST

In the discussions at SMORN-IV on the results of the physical benchmark test, it was
stated that comparison of different methods for determination of multivariate AR parameters
and checking of programming is imperative when more and more noise analysis groups tend to
use the AR method. Besides, several problems in the application of the method still remain,
among which determination of the "optimum" model order, choice of sampling time and inter-
pretation of results of an AR analysis in terms of system identification. The previous
benchmark test revealed that, although there was a reasonable agreement on parameters to be
determined from noise spectra like time constants or resonance parameters, the results
showed too large differences to conclude that established methods are available for deter-
mination of such parameters. Also, covariance matrices from intrinsic noise sources showed
differences of orders of magnitude.

The objective of the present benchmark test was to provide stationary analog signals from
a well-defined and relatively simple system, which exhibits several features of interest for
AR modelling and signals which contain certain anomalies during certain periods of time,
which anomalies should be found with appropriate anomaly detection methods, but should not
be obvious. Moreover, specific tasks are to be executed in order to test the capabilities of
the analysis methods applied and to stimulate further theoretical and experimental inves-
tigations on specific issues regarding AR modelling and anomaly detection.
The objective of the anomaly detection part of the benchmark test is not only to determine
the earliest instant a certain method detects the anomaly, but also to trace those parts of

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the system and/or the noise sources in the system that caused the anomaly and to raise questions about the false alarm rate of the method used.

3. SIGNAL GENERATION

The artificial noise data have been generated by the digital reactor core simulator BRASSDEL of the Delft Interfaculty Reactor Institute (IRI), which simulates reactor core dynamics with thermal hydraulic feedback. With its standard parameters it simulates the 2 MWth MTR-type reactor HPR at IRI.

The digital data from the simulator, calculated for successive time steps 6s=5 ms., were converted to analog data using a 12 bit DAC and were recorded on analog tape after filtering to obtain a smooth analog signal. Test signals were added before and after the noise data for calibration and definition of the time scale. One hour of stationary noise data was generated, followed by another one and a half hour of noise data with anomalies introduced, as specified in the next section.

This analog tape is considered as the primary data source for this benchmark test. A digital tape for distribution with the benchmark test was prepared at the Energy Research Centre at Petten, The Netherlands, by sampling the analog tape with 32 Hz sampling frequency, using appropriate anti-aliasing filters.

4. SYSTEM MODEL

In order to provide complete details of the system for anyone who might wish to analyse the benchmark test signals in the future, the system will be described in sufficient detail.

The model of the digital simulator is based on point kinetics with delayed neutrons and a simple thermohydraulics model with lumped fuel and coolant. Three signals were selected for this benchmark test, namely fuel temperature fluctuations, coolant temperature fluctuations and reactor power fluctuations.

Six groups of delayed neutrons were used with data from ENDF/B-IV (Garber et al., 1975) for thermal fission of U-235 and the same effective factor for the fraction of delayed neutrons for all delayed neutron groups. The point kinetic equations are solved in a prompt approximation with the zero power reactor transfer function for a linearized model and the continuous signal case included in Table I.

The heat transfer from fuel to coolant is described by the equation

$$C_f \frac{dT_f}{dt} = P(t) - \alpha(T_f(t) - T_c(t))$$

with $T_f$ and $T_c$ the fuel and coolant temperature, $P$ reactor power, $C_f$ the total heat capacity of the fuel and $\alpha$ the total heat transfer coefficient. This heat balance expresses the heating up of the fuel as the difference between the power produced in the fuel and the heat transferred to the coolant. The heating up of the coolant is balanced by the heat transferred from the fuel and removal by the streaming coolant, according to the equation

$$C_c \frac{dT_c}{dt} = \alpha(T_f(t) - T_c(t)) - 2C_c T_c(t)$$

with $C_c$ the total heat capacity of the coolant and $\tau_c$ the transit time of the coolant through the core. In the last term of this equation the difference between outlet and inlet temperature of the coolant have been set equal to two times the lumped coolant temperature. All temperatures are taken relative to the inlet temperature. Both fuel and coolant temperatures have a feedback to reactivity with constant reactivity coefficients $\rho_f$ and $\rho_c$ respectively:

$$\rho = \rho_f T_f + \rho_c T_c + \rho_o$$

with $\rho_o$ the constant reactivity from all other sources.

In the digital simulator the differential equations are represented by difference equations using backward differences for the derivatives. Table I shows the values of the parameters used in the simulation.

In order to generate noise signals three independent noise sources are introduced using pseudo random number generators providing Gaussian distributed random numbers. The first noise source acts on the fuel temperature and could be understood in physical terms as fluctuations in the coolant velocity, influencing the heat transfer coefficient (its direct effect on the coolant temperature has been neglected in order to keep the noise sources acting on the measured signals independent of each other). A colored spectrum for this noise source was obtained using a first order AR process according to

$$m_t = am_{t-1} + n_t$$
### Table I. System parameters and functions

<table>
<thead>
<tr>
<th>quantity</th>
<th>description</th>
<th>value</th>
</tr>
</thead>
<tbody>
<tr>
<td>$\delta t$</td>
<td>time step in digital simulation</td>
<td>5 ms</td>
</tr>
<tr>
<td>$\beta_{eff}$</td>
<td>effective fraction of delayed neutrons</td>
<td>0.0075</td>
</tr>
<tr>
<td>$1/C_f$</td>
<td>inverse total heat capacity</td>
<td>$15.4 \times 10^{-6}$ K/J</td>
</tr>
<tr>
<td>$1/\tau_f=\alpha/C_f$</td>
<td>inverse fuel time constant</td>
<td>$3.7 \text{ s}^{-1}$</td>
</tr>
<tr>
<td>$1/\tau_c=\alpha/C_c$</td>
<td>inverse coolant time constant</td>
<td>$0.8 \text{ s}^{-1}$</td>
</tr>
<tr>
<td>$1/\tau_{ce}=1/\tau_c \cdot 2/\tau_b$</td>
<td>inverse effective coolant time constant</td>
<td>$2.46 \text{ s}^{-1}$</td>
</tr>
<tr>
<td>$r_f$</td>
<td>reactivity coefficient of fuel</td>
<td>$2 \times 10^{-5}$ K$^{-1}$</td>
</tr>
<tr>
<td>$r_c$</td>
<td>reactivity coefficient of coolant</td>
<td>$-64 \times 10^{-5}$ K$^{-1}$</td>
</tr>
<tr>
<td>$P_0$</td>
<td>average reactor power</td>
<td>$2.09 \text{ MW}$</td>
</tr>
<tr>
<td>$\sigma_1$</td>
<td>standard deviation of noise source $N_1$</td>
<td>$0.1715\text{ K}$</td>
</tr>
<tr>
<td>$\sigma_2$</td>
<td>standard deviation of noise source $N_2$</td>
<td>$0.1661\text{ K}$</td>
</tr>
<tr>
<td>$\sigma_3$</td>
<td>standard deviation of noise source $N_3$</td>
<td>$0.00005\text{ K}$</td>
</tr>
<tr>
<td>$\alpha$</td>
<td>coefficient of AR filter of noise source $N_1$</td>
<td>$0.97$</td>
</tr>
<tr>
<td>$\alpha_a$</td>
<td>natural resonance frequency of filter of noise source $N_3$</td>
<td>$3.1416\text{ Hz}$</td>
</tr>
<tr>
<td>$\zeta$</td>
<td>damping factor of noise source $N_3$</td>
<td>$0.8$</td>
</tr>
<tr>
<td>$H_z(f)$</td>
<td>transfer function from power to fuel temperature</td>
<td>$\frac{1/a}{1+2\pi jf r_f}$</td>
</tr>
<tr>
<td>$H_3(f)$</td>
<td>transfer function from fuel to coolant temperature</td>
<td>$\frac{1}{1+2\pi jf \tau_{ce}}$</td>
</tr>
<tr>
<td>$H_4(f)$</td>
<td>transfer function from coolant to fuel temperature</td>
<td>$\frac{1}{1+2\pi jf \tau_{ce}}$</td>
</tr>
<tr>
<td>$S_1(f)$</td>
<td>shaping filter of noise source $N_1$</td>
<td>$1 \text{ for } f &lt; 8 \text{ Hz}$</td>
</tr>
<tr>
<td>$S_2(f)$</td>
<td>shaping filter of noise source $N_2$</td>
<td>$1 \text{ for } f &lt; 8 \text{ Hz}$</td>
</tr>
<tr>
<td>$S_3(f)$</td>
<td>shaping filter of noise source $N_3$</td>
<td>$\text{zero power reactor transfer function (prompt jump approx.)}$ $\frac{1}{1+(f/f_{r})^2+2\pi jf/f_{r}}$</td>
</tr>
</tbody>
</table>

with $m_r$ the discrete value of the noise component acting on the fuel temperature and $n_t$ a Gaussian distributed random number.

The second noise source causes fluctuations in the inlet coolant temperature. This noise source was obtained by filtering the Gaussian distributed random numbers of a second independent random number generator by a digital 5th-order Chebyshev filter (Gifford, 1985; 1986) with cut-off frequency at 8 Hz to obtain a band limited flat noise source spectrum. The third noise source models reactivity noise and was obtained by filtering the Gaussian distributed random numbers from a third independent random number generator by a digital second-order low-pass filter whose analog counterpart has the form

$$S_3(f) = \frac{1}{1-(f/f_{r})^2 + 2\pi jf/f_{r}}$$
with \( f \) the natural frequency and \( \zeta \) the damping factor. For the noise signals the block scheme of the simulator is given in Fig. 1. Most transfer functions are given in their analog form, although they were all applied in the form of their digital counterpart. The single step delays in the boxes with inscription \( z^{-1} \) are introduced because otherwise instantaneous transfer of information between pairs of output signals will take place, which prohibits any AR modelling (Hoogenboom et al., 1986; Ciftcioglu et al., 1988). Table I gives the values of the parameters used in the simulator. Fig. 2 gives the block scheme as assumed in an AR analysis, where a noise source is assumed to act directly on each measured signal.

![Block diagram of simulator](image)

Fig. 1. Block diagram of simulator

For the anomaly part of the data two anomalies were introduced. For the first anomaly the damping factor \( \zeta \) of the shaping filter \( N_a \) of the reactivity noise source (equivalent to the reactor power noise source in Fig. 2) was changed from 0.8 to 0.4, resulting in a weak resonance behaviour of this noise source with resonance frequency about 3 Hz, which might be interpreted physically as a vibrating control rod. This anomaly lasted 5.12 min, after which the original situation was restored. The second anomaly consisted of a linear change in time of reactivity coefficients \( r_p \) and \( r_c \) in such a way that mean reactivity remains constant. During a period of 28.33 min \( r_p \) changed from 2 to 24.75 pcm/K and \( r_c \) changed from -64 to -134.6 pcm/K (pcm=per cent mille=10^{-3}). Thereafter the system was kept in that stage, leaving about half an hour of stationary data from a slightly different system. The change of reactivity coefficients introduces a weak resonance in the overall transfer function from reactivity to power at about 0.6 Hz (Ciftcioglu, 1986). Fig. 3 gives the timing diagram for the anomalies.

5. TASKS TO BE PERFORMED

For the analysis of the stationary part of the signals the following transfer functions were requested in graphical form: \( H_2, H_3 \) and \( H_4 \), which may be obtained from a multivariate AR model. Also requested was the result of the AR representation of the open-loop transfer function from signal \( X_p \) (power) to \( X_c \) (coolant temperature), whose direct transfer function does not exist in the model. This task was included to test whether the AR analysis provides indications for physically non-existent open-loop transfer functions. For comparison and to hint at the distinction, the closed-loop (overall) transfer function from power to coolant temperature was also requested.

Furthermore, the autopower spectral densities of the noise sources as they act on the measured signals were requested in order to test whether AR analysis provides the true physical noise sources. Also requested were the relative noise source contribution ratios of each noise source to each output signal.

Quantitative determination of several time constants and gain factors of transfer functions or noise source spectra was requested. Finally the covariance matrix of the intrinsic noise sources as calculated by the AR analysis was requested.
With regard to the anomaly part of the signals the main question was to determine the earliest time instants and the duration of each anomaly in each signal. For each anomaly found determination was requested of the noise source and/or transfer functions that changed, causing the anomaly. Finally the false alarm rate for the method of anomaly detection was requested in order to point out the importance of this quantity and to see the supposed connection with a timely detection of an anomaly.

6. RESULTS

Contributions have been received from seven groups, listed in table II. Not all contributions were received in time for a thorough evaluation. Three participants used the analog data tape, four participants the digital data tape as the primary data source.
Table II. Participants in the artificial noise benchmark test

<table>
<thead>
<tr>
<th>group</th>
<th>country</th>
<th>names of participants</th>
</tr>
</thead>
<tbody>
<tr>
<td>Tampere University</td>
<td>Finland</td>
<td>H. Mustonen</td>
</tr>
<tr>
<td></td>
<td></td>
<td>H. Jokinen</td>
</tr>
<tr>
<td></td>
<td></td>
<td>J. Pohlus</td>
</tr>
<tr>
<td></td>
<td></td>
<td>P. Liewers</td>
</tr>
<tr>
<td>ZFK</td>
<td>GDR</td>
<td>E. Morishima</td>
</tr>
<tr>
<td></td>
<td></td>
<td>E. Türkcan</td>
</tr>
<tr>
<td></td>
<td></td>
<td>A.G. Federico</td>
</tr>
<tr>
<td></td>
<td></td>
<td>R. Regina</td>
</tr>
<tr>
<td>ECN</td>
<td>Netherlands</td>
<td>Y. Kuwada</td>
</tr>
<tr>
<td></td>
<td></td>
<td>H. Suzuki</td>
</tr>
<tr>
<td></td>
<td></td>
<td>K. Hayashi</td>
</tr>
<tr>
<td>ENEA</td>
<td>Italy</td>
<td>Y. Shinohara</td>
</tr>
<tr>
<td></td>
<td></td>
<td>J.E. Hoogenboom</td>
</tr>
<tr>
<td>Tokai University</td>
<td>Japan</td>
<td></td>
</tr>
<tr>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>JAERI</td>
<td>Japan</td>
<td></td>
</tr>
<tr>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>IRI</td>
<td>Netherlands</td>
<td></td>
</tr>
<tr>
<td></td>
<td></td>
<td></td>
</tr>
</tbody>
</table>

6.1 Stationary data

Most participants who analysed the stationary data used a multivariate autoregressive model to determine the system transfer functions and noise sources. Different criteria for model order determination were used, as well as different sampling times. Some analysis parameters are given in Table III, where each participant is designated by a letter, chosen at random.

Table III. Analysis parameters and conditions

<table>
<thead>
<tr>
<th>participant</th>
<th>A</th>
<th>B</th>
<th>C</th>
<th>D</th>
<th>E</th>
<th>F</th>
<th>G</th>
</tr>
</thead>
<tbody>
<tr>
<td>data tape</td>
<td>digital</td>
<td>digital</td>
<td>analog</td>
<td>analog</td>
<td>digital</td>
<td>analog</td>
<td>digital</td>
</tr>
<tr>
<td>analysis method</td>
<td>MAR</td>
<td>MAR</td>
<td>MAR</td>
<td>MAR</td>
<td>-</td>
<td>UAR</td>
<td>MAR</td>
</tr>
<tr>
<td>algorithm</td>
<td>L-D/a</td>
<td>L-D/a</td>
<td>L-D/a</td>
<td>L-D/a</td>
<td>-</td>
<td>L-D/b</td>
<td>LS-H/b</td>
</tr>
<tr>
<td>model order criterion</td>
<td>preset</td>
<td>FPE/c</td>
<td>AIC/d</td>
<td>DIC/e</td>
<td>-</td>
<td>AIC/d</td>
<td>AIC/d</td>
</tr>
<tr>
<td>time step ( \Delta t ) (ms)</td>
<td>62.5</td>
<td>125</td>
<td>20</td>
<td>10.1</td>
<td>31.25</td>
<td>31.25</td>
<td>31.25</td>
</tr>
<tr>
<td>filter freq. (Hz)</td>
<td>8</td>
<td>4</td>
<td>12</td>
<td>10</td>
<td>-</td>
<td>4.5</td>
<td>-</td>
</tr>
<tr>
<td>model order p</td>
<td>10/20/40</td>
<td>68</td>
<td>26</td>
<td>15</td>
<td>-</td>
<td>6-7-8</td>
<td>15-16-12</td>
</tr>
<tr>
<td>memory time ( \Delta t ) (ms)</td>
<td>625</td>
<td>8500</td>
<td>520</td>
<td>626</td>
<td>-</td>
<td>-350</td>
<td>-450</td>
</tr>
</tbody>
</table>

\( ^a \)Levinson-Durbin  \( ^b \)Least-squares with Householder transformation  \( ^c \)Final Prediction Error  \( ^d \)Akaike's Information Criterion  \( ^e \)Gradient of AIC

Five participants produced the requested graphs of transfer functions, noise source spectrum and noise contribution ratios. Besides problems of some participants with normalization to the requested physical units, comparison of the results for the transfer function \( H_3 \) showed reasonable agreement, while the results for \( H_2 \) and \( H_4 \) showed appreciable differences, both in modulus and phase behaviour. Participant A produced results for model orders 10, 20 and 40 with considerable differences in \( H_2 \) and \( H_4 \), especially at low frequencies, although the noise source covariances were already converged at the lower model order.

Results for the physically not existing transfer function \( H_{32} \) from power to coolant temperature differed even more. No indications were given to conclude that this transfer function in fact was not present in the system. From a comparison with the overall transfer function from power to coolant temperature as obtained by participant C the open-loop transfer function was lower by two orders of magnitude, which might be used as an indication for negligible direct signal transmission.

The forms of the auto power spectrum of noise source \( Q \), as obtained by the participants resemble each other quite well, except for normalization and with the exception of the low frequency behaviour of the result of participant B. The noise contribution ratios to the reactor power signal show good agreement. Those for fuel temperature and coolant temperature showed reasonable agreement. Most participants calculated the NCRs in such a way that they always add up to one exactly.
From the transfer functions and the noise spectrum obtained, several numerical values were to be determined. The results together with the theoretical values are shown in Table IV. Not all participants could determine all requested quantities, as the functions from which these quantities were to be obtained did not behave as assumed in the benchmark test description. As can be seen from this table several quantities show reasonable agreement, in particular the parameters of transfer function $H_2$ and the parameters from $H_3$ as far as participants C and D are concerned. The parameters of the noise source $Q_1$ are poorly estimated. The elements of the variance-covariance matrix of the residual noise show large deviations. Even the off-diagonal elements of the normalized covariance matrix show distinct differences.

**Table IV. Numerical results for stationary data**

<table>
<thead>
<tr>
<th>function quantity</th>
<th>units</th>
<th>theoretical</th>
<th>participant</th>
</tr>
</thead>
<tbody>
<tr>
<td></td>
<td></td>
<td>theoretical</td>
<td>A</td>
</tr>
<tr>
<td>$H_2(f)$</td>
<td>$A_2$</td>
<td>$k/MW$</td>
<td>$1/\alpha=4.16$</td>
</tr>
<tr>
<td></td>
<td></td>
<td></td>
<td>$\tau_2$</td>
</tr>
<tr>
<td>$H_3(f)$</td>
<td>$A_3$</td>
<td>-</td>
<td>$\tau_{ce}/\tau_c=0.325$</td>
</tr>
<tr>
<td></td>
<td>$\tau_3$</td>
<td>$\tau_{ce}$=0.407</td>
<td>0.41</td>
</tr>
<tr>
<td>$H_4(f)$</td>
<td>$A_4$</td>
<td>-</td>
<td>1</td>
</tr>
<tr>
<td></td>
<td>$\tau_4$</td>
<td>$\tau_f=0.270$</td>
<td>-</td>
</tr>
<tr>
<td>$Q_1(f)$</td>
<td>$C$</td>
<td>$k^2/Hz$</td>
<td>$2\delta_{11}a=3.16 \times 10^{-4}$</td>
</tr>
<tr>
<td></td>
<td>$\tau_5$</td>
<td>$\delta_{11}$=0.162</td>
<td>0.20</td>
</tr>
<tr>
<td></td>
<td>$\tau_6$</td>
<td>$\tau_f=0.270$</td>
<td>0.22</td>
</tr>
<tr>
<td>$Q_1(f)$</td>
<td>$\tau_7$</td>
<td>$\delta_{11}$=0.162</td>
<td>-</td>
</tr>
<tr>
<td>$\Sigma$</td>
<td>$\sigma^2_{11}$</td>
<td>$\nu^2$</td>
<td>4.62 $10^{-3}$</td>
</tr>
<tr>
<td></td>
<td>$\sigma^2_{12}$</td>
<td>$\nu^2$</td>
<td>6.67 $10^{-4}$</td>
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<td></td>
<td>$\sigma^2_{13}$</td>
<td>$\nu^2$</td>
<td>1.74 $10^{-2}$</td>
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<td>$\sigma^2_{22}$</td>
<td>$\nu^2$</td>
<td>5.13 $10^{-3}$</td>
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<td>$\sigma^2_{23}$</td>
<td>$\nu^2$</td>
<td>-4.91 $10^{-3}$</td>
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<td>$\sigma^2_{33}$</td>
<td>$\nu^2$</td>
<td>5.21 $10^{-2}$</td>
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*aafter normalization to physical units where appropriate*
6.2 Anomaly data

Analysis of the anomaly data was given by participants A, B, D, E and F, each using a different method for anomaly detection. Table V summarizes the most important results.

Participant A used the residual noise obtained from the actual signals after subtracting the value predicted by the AR model determined for the stationary data. This led to a quite accurate detection of the first anomaly and a number of short sharply peaked anomalies from at most a few consecutive data points, two of which during the stationary signal period. These deviations were not present in the analog signals and must have been introduced during the sampling process to obtain the digital data tape. Moreover, the first decay constant of the correlation function determined from short (64 s) data segments was used as an anomaly indicator, from which not only the first anomaly was detected, but also the second one, using trend analysis, although its start could not be determined accurately. Extensive studies were made from the last part of the signals where the second anomaly is stationary.

Table V. Results of anomaly detection

<table>
<thead>
<tr>
<th>participant and method</th>
<th>anomaly number</th>
<th>signal</th>
<th>start (s)</th>
<th>end (s)</th>
<th>system changes</th>
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<tr>
<td>anomalies actually present</td>
<td>1</td>
<td>$(X_1, X_2) X_3$</td>
<td>4039</td>
<td>4346</td>
<td>$Q_3$</td>
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<tr>
<td></td>
<td>2</td>
<td>$(X_1, X_2) X_3$</td>
<td>5555</td>
<td>7254</td>
<td>$H_{13}, H_{23}$ (slowly varying)</td>
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<tr>
<td></td>
<td></td>
<td>$(X_1, X_2) X_3$</td>
<td>7254</td>
<td>end of signals</td>
<td>$H_{13}, H_{23}$ (constant)</td>
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<tr>
<td>A residual noise</td>
<td>1, 2, 4-15</td>
<td>$X_1, X_2, X_3$</td>
<td>duration &lt;0.2 s</td>
<td>4062.3</td>
<td>4329.7</td>
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<tr>
<td></td>
<td>3</td>
<td>$X_3$</td>
<td>4030</td>
<td>4350</td>
<td></td>
</tr>
<tr>
<td>change of decay time constant of correlation function</td>
<td>1</td>
<td>$X_3$</td>
<td>4030</td>
<td>4350</td>
<td></td>
</tr>
<tr>
<td></td>
<td>2</td>
<td>$X_3$</td>
<td>4350</td>
<td>7038</td>
<td></td>
</tr>
<tr>
<td></td>
<td>3</td>
<td>$X_3$</td>
<td>7038</td>
<td>end of signal</td>
<td>$Q_1, Q_3, H_{13}, H_{21}, H_{23}$</td>
</tr>
<tr>
<td>B change of system modelling parameters</td>
<td>1</td>
<td>$X_3$</td>
<td>5790</td>
<td>end of signal</td>
<td></td>
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<td>D short time FFT spectra</td>
<td>1</td>
<td>$X_3$</td>
<td>3800</td>
<td>4300</td>
<td>$Q_3, H_{31}$</td>
</tr>
<tr>
<td></td>
<td>2</td>
<td>$X_2$</td>
<td>4300</td>
<td>4500</td>
<td>$Q_2$</td>
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<tr>
<td></td>
<td>3</td>
<td>$X_3$</td>
<td>5900</td>
<td>6250</td>
<td>$Q_3, H_{31}, H_{12}$</td>
</tr>
<tr>
<td></td>
<td>4</td>
<td>$X_1$</td>
<td>6500</td>
<td>6800</td>
<td>$Q_1$</td>
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<tr>
<td>E short time variance</td>
<td>1</td>
<td>$X_3$</td>
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<td>4336</td>
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<tr>
<td></td>
<td>2</td>
<td>$X_3$</td>
<td>5631</td>
<td>8284</td>
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<tr>
<td>F short time AR parameters</td>
<td>1</td>
<td>$X_3$</td>
<td>4001</td>
<td>-</td>
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</table>
Participant B used the behaviour of certain modelling parameters from the AR process to detect possible anomalies and was able to detect the second anomaly almost from its start.

Participant D used short time FFT spectra for anomaly detection. From this the first anomaly was detected from an increase in the reactor power noise around 3 Hz. Some other anomalies found, however, are false.

Participant E calculated (amongst others) the variance of the signals for every 8 s signal duration. From a graph both anomalies could be detected quite accurately.

Participant F used AR modelling for short data segments and studied changes in the first few AR parameters. From a graph the start of the first anomaly could be detected. However, there might be an error in his definition of zero time. Further interpretation of the results with regard to the duration of this anomaly and the presence of the second anomaly was difficult.

Participant G detected an anomaly, but no specific results about the onset and duration of the anomalies were reported.

7. CONCLUSIONS AND RECOMMENDATIONS

From the results of the analyses of the stationary data it can be concluded that no satisfactory agreement consists on the results of the AR modelling of the present system. It is not yet possible to state whether the programs used by the participants are on their own correct and deviations are caused by different choice of parameters like sampling time and model order.

Most participants emphasized the relatively large covariances between different noise sources (off-diagonal elements of the residual noise covariance matrix) and suggested that the noise sources must be correlated. However, this was not the case and the correlation found must be due, in the opinion of the present authors, to the use of a too low sampling frequency, which must be much higher than inferred from the frequency band of the signals. Therefore, re-analysis of the data with higher sampling rate is recommended. To this end use of the analog data tape is recommended for a free choice of sampling time. In case a digital tape is necessary, a new digital tape with much smaller sampling time is needed.

Different criteria for model order determination were used. Except for participant B, the memory time of the AR model (model order times time step) was in reasonable agreement with each other. Nonetheless further investigations in optimum model order is recommended.

Normalization to the requested physical units gave problems to several participants. As this is of importance for a comparison of absolute data, more attention should be given to normalization. No use was reported of the test signals preceding and following the noise data on tape for calibration of the signal amplitude and time scale.

Large deviations were found in the calculated covariance matrices of the residual noise source of each signal. Although these values depend on the sampling time used, other errors or misinterpretation seem to be a cause. A closer look at the calculation of the variance of residual noise, given the AR coefficients, is needed.

With regard to the anomaly detection, quite different methods were applied. In principle both anomalies could be detected, but not every method applied was capable of detecting each anomaly. Using a simple means as calculation of variance of the time signal over a relatively short period turned out to be sufficient for detection of both anomalies. However, the second anomaly could only be detected from trend analysis of some system parameter, i.e. after judging parameter behaviour over a longer period, well extending beyond the onset of this anomaly. The present authors have investigated several methods based on the residual noise obtained after subtraction of the AR prediction of the signal value, as was also used by participant A for the detection of the first anomaly. With further refinements of the method it was possible to detect both anomalies on-line at an early stage.

As a general conclusion, it can be stated that the present benchmark test was useful in revealing current problems connected to the application of AR modelling and anomaly detection. Its usefulness will increase if more participants will take part and the present experience is fed back to the participants for reevaluation of their results and application of refined anomaly detection methods. Comparison of results from separate steps in the AR determination will provide a better check on the consistency and correctness of the computer programs used. Further theoretical work is recommended with regard to choice of sampling time and selection of AR model order, detection and treatment of physically non-existing open-loop transition matrix between noise source covariance and the actual noise source strength. Application of fitting procedures to complex functions as system transfer functions needs more attention. Information is needed on the statistical accuracy of the AR coefficients and the transfer functions obtained, which information may be used in determining weight factors in the fitting procedure.
REFERENCES


CONCLUDING SPEECH
CONCLUDING SPEECH

W. BASTL (F.R.G.)

Summary of the Discussion

Let me now draw the conclusions from our panel discussion. At first we found out that aging is an issue where we noise people can contribute substantially with our monitoring methods. As aging is very important, our plants become older and older, this is a challenge for us. In addition I might mention also longer fuel cycles of the plant which is a tendency nowadays because of economic reasons, and also load following operation. These are certainly also areas where we can contribute with our methods. Then there was the observation that we should think more about combining — let me say — the ordinary (mean value) measurements with our noise analysis measurements, that is to combine these two pieces of information and get out even more useful information. And there was the conclusion that noise analysis evidently gets more and more adapted by the users, by the operators. In this context SMORN V was found to be a breakthrough because of the substantial high contribution and participation from utilities. There was the question on standards. Certainly, some national standards have been developed already, but we have to go on in this area. I like to mention here that in the International Electrical Commission, IEC, there is a standard on loose parts monitoring on the way, and a standard on vibration in preparation. We also found out that — and this was in connection with the benchmark testing — that some of the methods need to be further validated, others need to be further developed. You do know there are many new ideas on more sophisticated analysis methods, and to validate these against practical cases, so as to be sure that they work correctly in industrial application, is of course very important. This was to me the main outcome of our discussion.

Of course it is impossible within a symposium like this to treat all the problems. It is very often overlooked that noise analysis — though it seems to be a very narrow specialists' field — has a very wide range of application. For that reason we had to focus on some major topics and could not really draw conclusions which covered all aspects.

Next Symposium

Turning over to the question of continuing our symposium series, I think so many problems are still to be solved in our field that there can be only one answer, "yes". Looking from outside into noise analysis one might get the impression that after working so long a time in a narrow field the methods should be matured, the R&D work should be finished. In responding we have to state that still a lot of work needs to be done and the potential of our methodologies is not at all exhausted. This is not at least underlined by the progress in the field since SMORN IV. And here I recall statements made at the end of SMORN IV, in being afraid that there would not be so much of a progress possible as to have a conference in the near future. I guess we all were surprised about the tremendous new work which has been done during the last years. I am confident this will progress that way because we just entered some of the applications. And of course new applications will call for new solutions.

Summary of Symposium

I now try to make some conclusions which you please should take as my personal ones. Thinking backwards a little bit about the SMORN meetings, we started off with basic research work, at the most. And then more and more practical applications came up, and nowadays we have to show really very important practical applications. I mention just loose parts, vibration, leakage monitoring.
It seems to me that there are two main branches of application in our field, namely structural or mechanical monitoring on the one hand and process monitoring on the other hand. I would say up to now we have advanced further with the structural part. But I do see a high potential in the process branch and in this context I like to mention the excellent work - also application work, though it is in the beginning - done by our Japanese colleagues. Since SMORN IV if not III, we have had separate sessions on safety applications. This was not only because of the fact that the Committee on the Safety of Nuclear Installations has been called in as a co-organiser of our meetings but because already at those times it was felt noise analysis methods can be very well used to solve safety issues. Nowadays we got a quite good mixture in our conference, in between application issues and R&D work, and I like to stress this is not the case because of organizing the symposium this way; I think it reflects the present situation.

Let me now come to a brief analysis where noise analysis now stands in my opinion. As to methodologies I do feel that a lot of work has to be done with respect to higher order correlation methods, to pattern recognition, to the problem of multiple input output system analysis and, to take a more specific example, in autoregressive modelling. The capability of AR to analyse cause and consequence effects, to analyse feedback systems, is extremely important. But as we have heard, there are still a lot of problems to be solved, before we can enter into practical application. Let me mention here just the problem of optimal selection of independent model parameters.

And now to the applications. As already mentioned to you, we have excellent applications with respect to monitoring structural parts in our plants. But this is, as a matter of fact, not really about on-line systems, it is rather about on-site systems. In the future we have to think more about the ability of our methods for on-line systems and more over to perform on-line process analysis. This means to me - in the framework of nuclear power plant instrumentation and control - we have at hand to contribute substantially in going from simple monitoring to diagnosis systems, and we can include here also prognosis. Thinking about this we have to realize that a revolution in I+C techniques as a whole is beginning which goes along with the new technologies in this field. I am talking about the very advanced electronic systems we already have or we will have in the near future; to use a direct translation from German: we say "Processors integrated I+C". This means the hardware will be there which can run all the programs used in noise analysis, even complicated ones. If we are able to industrialize, to customize and to validate our software - and this is a different situation than in former days - there is basically no limitation to applying all our methods in the I+C systems of the future. So it seems to me that the main direction we should go with our techniques - besides the specific issues we found out - could be what I would call on-line diagnostics, interpreting this term as the upcoming new methodology to be applied in future I+C systems. That is just about all my conclusions, where we should go in the future and where we have to go in the future.

Closing of the Symposium

Let me now come to the official closing of the Session and the Symposium. I should like to thank on behalf of OECD, on behalf of the IAEA and moreover on behalf of the Organizing Committee, our Session Chairmen for their excellent work and specifically our Benchmark Group, which did a tremendous job. I should not at all forget the excellent interpreters; it is not at all easy to translate our noisy terminology. Further I should like to thank our technical officers and all the support we got from the technical staff of the patent office. I should also not forget our friendly ladies at the conference desk and of course all the administration staff who took care of the social events. At the end I like to specifically thank my colleagues in the Organizing Committee for their stimulating cooperation.

And now I slip into my GRS suite and thank OECD for giving us the opportunity to organize SMORN V. I am grateful for all the support they gave us. Special thanks are due to the IAEA for the assistance in encouraging non-OECD member states to send delegates to our meeting. This was the main idea as to take in the Vienna Agency as co-sponsor; and I guess it has worked out quite well because there was an increased participation of those countries, specifically when considering their most valuable contributions. But at the end I should not forget one thing which is the major parameter for a successful conference, namely the papers sent in. This is of course something you have not in your hand as an organizer. I have to state here, and this is not only my statement, that we achieved an exceptional high standard of presentations. Therefore I have to thank all the contributors for their excellent work. This brings me now close to the end of this Symposium. I do wish the people participating in the technical visits next week, a nice weekend here in Munich and an interesting technical tour. I do wish all of you a good trip back home, and I hope we will see you all in three years from now in hopefully the same configuration, maybe with some more young people, at SMORN VI in the United States. Thank you very much again for coming to Munich. SMORN V is closed.
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