CSNI Report No. 43
Volume 1

SPECIALIST MEETING ON REGULATORY REVIEW IN THE LICENSING PROCESS

Madrid, Spain

7-9 November 1979

Hosted by the

JUNTA DE ENERGIA NUCLEAR

PROCEEDINGS

April 1980

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, Ireland, 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 objectives of NEA remain substantially those of ENEA, namely the orderly development of the uses of nuclear energy for peaceful purposes. This is achieved by:

- assessing the future role of nuclear energy as a contributor to economic progress, and encouraging co-operation between governments towards its optimum development;
- encouraging harmonisation of governments' regulatory policies and practices in the nuclear field, with particular reference to health and safety, radioactive waste management and nuclear third party liability and and insurance;
- forecasts of uranium resources, production and demand;
- operation of common services and encouragement of cooperation in the field of nuclear energy information;
- sponsorship of research and development undertakings jointly organised and operated by OECD countries.

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

The 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 the nuclear licensing process. 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 organise international co-operation in nuclear safety. 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 licensing authorities and to the scientific community al 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.

CONTENTS

VOLUME 1

PAGE

PAGE

PAGE

VOLUME 3

Abstract

The Specialist Meeting on Regulatory Review in the Licensing Process to be sponsored by the Committee on the Safety of Nuclear Installations (CSNI) of the OECD Nuclear Energy Agency was held in Madrid, Spain, from the 7th to the 9th November 1979. It followed up another CSNI Specalist Meeting on Regulatory Inspection Practices, and was hosted again by the Junta de Energía Nuclear.

The meeting was attended by representatives of sixteen countries and two international organizations.

The purpose of the meeting was to examine current regulatory review practices in member countries, and to draw up further considerations after the accident at TMI-2.

In all, 34 papers were presented to the meeting, covering the following topics:

- Organizational matters.
- Technical bases for the regulatory review.
- Scope and stages of the regulatory review.
- TMI and its impact on the licensing process.
- Technical methods of review and assessment.
- Relationship between research and regulatory review.
- Specific aspects of regulatory review.

The meeting proved to be very useful for discussing licensing practices and showed that regulatory review and its methodology are very similar everywhere. Different positions of regulatory bodies on special technical problems were also considered, and above all regulatory recommendations to improve nuclear safety were discussed after the Three Mile Island accident.

SESSION I

OPENING OF THE SPECIALIST MEETING

Chairman:	A. Alonso
Opening Authorities:	Tte. Gral. Olivares F. Pascual K.B. Stadie R.T. Kennedy P. Giuliani

From right to left
- K.B. Stadie
- F. Pascual
- Tte. Gral. Olivares
- R.T. Kennedy
- A. Alonso
- P. Giuliani

From left to right
- Alonso
- Kennedy
- Olivares
- Pascual
- Stadie

Opening Remarks

A. Alonso

Director: Nuclear Safety Department, Junta de Energía Nuclear

Mr. President, Authorities from the Nuclear Energy Agency, the U.S. Nuclear Regulatory Commission and the Spanish Junta de Energía Nuclear. Distinguished representatives from the various countries and International Organizations. Ladies and Gentlemen:

It is for me a great honour to be Chairman of the meeting we are just going to open on Regulatory Review in the Licensing Process. It is only a little over two years since we celebrated our meeting on Regulatory Inspection Practices, of which our meeting today should be considered a second logical part.

It is evident that the use of nuclear power to produce electricity is finding difficulties all over the world, and it is also true that nuclear safety is at the center of all this. That is the reason why we are having here in Spain a week full of events within our Organization.

In the early Seventies, the Nuclear Community was shattered to its very roots by the effectiveness and, on occasion, the bitterness of the opposition of our society to the use of nuclear energy. It was then clear that governments should react, and they did indeed react, by establishing the appropiate Bodies, or increasing the existing ones, to regulate nuclear activities deeper and better with public safety as a major aim.

It was within the above mentioned context that the Subcommittee on Licensing within the CSNI was created. Among the most important activities undertaken by the Subcommittee I must mention the organization of a Specialist Meeting on Inspection Practices which took place here in Madrid in September 1977.

This year, 1979, with the problem of public acceptance unresolved, the nuclear community was again shattered by the accident at Three Mile Island. Again the governments reacted, reanalyzing the Bodies they created to regulate nuclear activities. This fact will contribute to making our Specialist Meeting on Review in the Licensing Process an interesting one.

Before I finish let me express my deep appreciation to those who have contributed to the preparation and conduct of the meeting. In particular, the speakers and participants coming from foreign countries, the Members of the Working Group who designed the contents of the meeting and the members of the J.E.N. who are helping us in the conduct of our Sessions.

Let me finish by wishing you a very pleasant stay among us in every respect. Authorities, ladies and gentlemen, thank you for your attention.

Madrid, October 7th, 1979.

Invited Lecture

R.T. Kennedy

NRC Commissioner

Thank you, Mr. Chairman. Authorities, ladies and gentlemen: It is a very great pleasure and a great honour for me to have this opportunity to talk to you at a particularly fortuitous moment for me. I was delighted to learn that my visit to Spain for other matters happily coincides with the opening of this meeting, and I was honoured when I was asked if I might join here in the opening and say a few words about our own interests and concerns in nuclear safety.

Surely, if I had been before you a few months ago, maybe not so long ago, I think I would have been speaking about rather different things. We do and we must see things very differently in the light of Three Mile Island. It makes a world quite different for us than the one we thought existed last March. But it need not be a world so different so long as we keep in our minds the essential message that it conveys: that safety is our most important product. I know that, as the Chairman suggested, two years ago you had a most successful meeting and this meeting, as Mr. Chairman indicates, is a natural follow-up, a natural second step arising from that meeting. And this meeting is being held even though some members had believed that it might well wait a little longer. But Three Mile Island has perhaps made further considerations on safety matters an even more urgent consideration than earlier thought.

Two elements - safety analysis and inspection - are the two essential aspects of any nuclear safety programme, and it's only fitting that you should devote your time to sharing your experiences in these fields and fully develop practical working concepts in both of these fields. The safety analyses which you develop in concept must not be biased by assumptions about the type of government and utility organizations that will exist. It's important that the procedures adopted for each country take the proper account of the staffing, the division of safety authority and responsibility and other social and political institutions of the particular country. For example, in the United States we have an independent National Regulatory Commission, a body which has something of the order

of perhaps nearly 3,000 people. That's more full time persons working in nuclear matters than perhaps the total of people actually involved in the utilities who operate those plants.

Other countries with smaller nuclear energy programmes and a relatively greater number of utility safety experts must look to ways to divide up the work - and there's plenty of work to go around, as we all know - for ways to divide up the work in such ways that the important safety analyses actually get done. The important point is that the analyses must be performed. And the concepts that you develop for getting that work done must be flexible enough to adapt to the various conditions in the various countries where those concepts will be applied. I'm pleased to know that the United States will be presenting three papers at this meeting, and I hope that they will make a major contribution. I know those who have prepared them and will be delivering them, and I can assure you that they are of the highest order of capability and competence and good will.

If I could, I'd like to speak briefly, just very briefly, about two things, two subjects that I believe are worth your very careful consideration and the deliberations of your meeting. First, the need for expanded international reporting, analysis and use of conclusions which are drawn from reactor operation experience around the world. And the second, the economic value of safe reactor operation.

As to the first: our safety programs, all of us together, are interdependent, and they must be. As in many other endeavours of this very complex society in which we all live, we can succeed in protecting the public health and safety and advancing the public welfare and good only if we combine the best efforts of all. One of the most useful messages, I think, that have come from the Three Mile Island experience is the importance of making full use of all available reactor safety information and experience to reduce the possibility of a repetition of errors of design and operation.

One serious criticism that has been levied against the NRC - and there have been many - or one of the more serious, in my view, was that neither the industry nor the NRC paid sufficient attention to several clear precursor events - reactor transient incidents that, if they had been heeded, and if

they had been recognized for what they were, could possibly have helped to prevent the Three Mile Island accident or at least have substantially mitigated its effects. That is an indictment of a system, for if we cannot learn from what we know, and if we fail to recognize and make use of what we know, it is unlikely that we will have a safety program worth the name.

We already have, in the NRC, as I'm sure many of you know, a very elaborate system for reporting, collecting and studying information about incidents in the nuclear reactor operations, and for analyzing that data and, if needed, providing information, giving special warning of potential problems to utilities that are operating nuclear reactors of the type in which the incident has arisen. Despite all that system, and despite the several incidents which should have warned us, we and the industry in the United States did not react fast enough to prevent the accident.

And I would have to say that the problem extended beyond our shores. For at least one incident had occurred in Europe that should have contributed warning information to us. That European incident included some of the same equipment failures that occurred at TMI, and a warning that the coincident ECCS initiation logic which is common to Westinghouse but not to B&W PWR's should have been changed. That warning was not heeded, despite the European incident, partly because the European incident, though it had been reported in international power reactor operation reports, either never had been reported directly to the NRC, or picked up by the NRC from power outage reports and then used by the NRC.

The failure was ours. But it was a failure of the system too. For one can hardly expect, and I think this applies to all of us, one can hardly expect that all of the literature and the vast amount of reporting that goes on every day can in fact be reviewed totally effectively. It is asking more than the human mind can do. We need a better system for identifying, pinpointing and acting upon essential information. We're trying to make a better use of the information we have, and also we're trying to make it possible to be sure that more and more information is available to everyone. We need better reporting on our side, and from foreign reactor experience as well. We're going to continue to share our conclusions with the entire international community and plans are being made now, and I think this is encouraging, to exchange reactor safety experience information under the auspices of this organization, the CSNI. We're participating and encouraging this planning, and we'll support it.

On Monday this week I had the very great pleasure, and honour, to join with Chairman Olivares of the JEN in signing a five-year extension to the agreement which our two agencies have to continue the exchange of operating experience reports and conclusions that we reach about the significance of those reports. We have a whole lot of these, many such bilateral agreements, and with many of you here in this room, over 20 of them in all. It's important in our view, we hope in yours, that that sort of cooperation is expanded, broadened, to cover everyone who is in this business.

Now to the second point, the economics of reactor safety. That's a strange subject. Often it is considered irrelevant to speak of economics and safety in the same breath. But let me just suggest that a utility company, whether it be public or private, which is operating an investment which in our country may be a billion dollars for a single thousand megawatt device, a utility which does not consider that safety is its most important product, may in fact be jeopardizing the very economic life on which it depends. And if it does, it is also jeopardizing the welfare, not just the health and safety, but the ultimate welfare of those people which that utility serves.

Now that's a strange, perhaps an even revolutionary notion. But those who argue, and there are some, that the concept of economic operation and safety are not necessarily compatible are wrong, in my view. And I commend to you some thinking about where sensible management drives a well-managed utility. For it cannot be imagined that the investors, the stockholders of Three Mile Island, were served very well. They have lost a half billion dollars, and probably more. We're obviously very concerned about Three Mile Island in every aspect. How close a near-miss accident was this? We'll debate that and take every reasonable step to avoid any future danger of this kind to the public. But, I submit, the utilities ought to be thinking too that if the balance sheet suggests to them that unless a regulatory agency demands a safety step it will be unnecessary to pursue it, a utility is making in my judgement a great management mistake. It is making a decision which is unsound. For if it sees a way that its plant can be operated more safely, if the training of its operators can be improved to increase the likelihood that they can operate that plant under the finest of conditions, and with the ultimate of safety, and they do not take that action, they are making a great mistake not only in the interest of public health and safety, but in their own economic interest. That's

bad management. There was a time when bad managements were replaced. Well, let me just suggest one last thesis about that point.

It is being said often now that because of Three Mile Island, that safety is after all the responsibility of the regulator. And I say that's false. Safety must begin with the person who conceives a nuclear plant. It must be paramount in the minds of those who build it. And it must never leave the consciousness of those who operate it. Only then, and only in those circumstances, can regulation contribute significantly and importantly to safety. Because if those who are building and operating plants do not see that their most important product and the one thing that must drive them is safety, then all the work that we as regulators may do will be following; we cannot lead them. We can help them, we can look over them; we cannot lead them. They must lead themselves. It must be a matter that is so important to them that they conceive what we do as regulators as help, not limits.

To summarize, then, very briefly, I believe it's essential that, first, we increase our international cooperation in reporting and using reactor operating experience. I know that you all share this view. And secondly I think that we must try our very best to encourage throughout the industries, be they public or private, which operate and construct these plants, we must encourage them to think hard, harder, ever harder, about safety, not just operating a plant, because it is not just another water boiler to produce steam to turn a turbine. It is a very complex, and very efficient, but a very complex machine. That's the message they have to get.

We're all most indebted to Dr. Alonso for his superb efforts. We appreciate the work that the Junta has done and the cooperation it has given us. We pledge that cooperation to it, and we know that they worked very hard to develop this meeting, and on its organization. We congratulate them and you. We congratulate Mr. Stadie of the NEA for his work on the NEA side, and wish you a very very happy and successful visit here in Madrid. I hope it will be as pleasant and successful, indeed, as my own has been. Thank you very much. It was good to see you.

Introductory Remarks

K.B. Stadie

Head, NEA Nuclear Safety Division

Chairman Olivarez, Commissioner Kennedy, Dr. Pasquel, Dr. Alonso, Ladies and Gentlemen,

It is a great pleasure for me to be back here in Madrid and to welcome you to the CSNI Specialist Meeting on Regulatory Review in the Licensing Process. This is the second specialist meeting sponsored by the CSNI Sub-Committee on Licensing. We met here in Madrid two years ago to discuss regulatory inspection practices and the success of that meeting encouraged us to be more ambitious and try to tackle the much broader question of regulatory review.

Meetings of this kind, with discussion in detail of the problems met by regulatory authorities, are a recent innovation in the CSNI programme. In a wider sense, however, one major aim of CSNI has always been to facilitate your task of assessing applications for building and operating nuclear power plants. This may seem a contradiction to you, since the major thrust of CSNI's activities is in safety research. However, we have always seen the essential objective of this programme as attempting to ensure that the people who license reactors do in fact benefit from all the results of the safety research undertaken by the OECD countries - on which they are currently spending $1 billion a year - and from the operating experience from the over 150 plants in service.

Beyond this, CSNI is trying to bring about international consensus on key safety issues, which is also likely to help the acceptance of nuclear power and, as a result, your task.

Mr. Chairman, rather than dwell further on the relevance of the CSNI programme in licensing, I should like to tell you about our plans and concerns for international co-operation in safety research and licensing between OECD countries.

For obvious reasons, OECD activities in nuclear safety have increased rapidly over the past five years and now form the largest part of the NEA programme. In developing this programme since 1974 the Committee on the Safety of Nuclear Installations has identified a great number of questions which could profitably be followed up internationally, in addition to the limited number of questions which we are actually able to deal with. We have continued nevertheless to give warning that a nuclear accident anywhere would be an accident everywhere. Regrettably, TMI has proved our point. The special meeting of CSNI in June heard that nearly all countries with nuclear power are now thoroughly reviewing their safety programmes; in fact the impact of the accident on nuclear power development elsewhere may be even more severe than in the United States.

The case for close international co-operation is now stronger than ever, not only to make another TMI less likely but also to avoid decisions about more technical and regulatory fixes being taken hastily without ample justification. In this context the CSNI, at its meeting next week, will consider a number of proposals aimed at producing an international consensus, or better, a common position.

In addition, it is our view that increased international co-operation in nuclear safety research should quickly resolve the outstanding safety questions, especially the problems highlighted by the TMI accident. We see this as a logical expansion of the CSNI programme, particularly in those areas in which the Committee has pioneered new kinds of international co-operation. These cover information systems and data banks, standard problem exercises, interlinking of national safety research facilities and specific state-of-the-art reports.

Let me say a few words about these four principal areas of international co-operation and how we see these activities developing in the light of TMI.

One of the most natural tasks of an international organisation is to collect and collate technical data and information. This is particularly relevant to CSNI because of the enormous increase in the volume of results from the $1 billion/year safety research programme in the OECD area. You are all familiar with the Nuclear Safety Research Index which this year lists 1400 individual projects. More recently we have begun to develop an information system which is intended to report operating experience from 150 plants in service in our Member countries. Its value in the context of the TMI accident is obvious. Over the past year a special CSNI Group of Experts has set criteria for selecting reportable events. There are of course some legal and administrative hurdles which at the moment prevent certain OECD countries from participating in the information system. Early next year OECD will convene a meeting of governmental representatives in order to overcome these obstacles. In addition to these exchanges it seems to us highly desirable to arrange a more systematic collection of technical data about the reliability of systems and components, with a view in particular to enriching the data base for quantitative risk assessment.

You will be familiar with the CSNI international standard problem exercises. Here we are trying to gauge the reliability and relative accuracy of the different kinds of assessment tools you, the regulators, need to assess the safety of nuclear installations. These tools include computer codes, measurement techniques or simply, instruments, methods of testing materials, etc.

So far CSNI has pioneered this kind of international standard problem exercise in several specific areas, notably for computer codes predicting LOCA-ECCS behaviour and containment response, as well as for non-destructive testing of welds in heavy steel sections using ultrasonics. The Committee has identified some 20 other applications for ISP exercises. In our opinion, they provide the most efficient way of increasing your confidence in these assessment tools and they should be taken up speedily.

To be frank we have been less successful in the third area, the interlinking of national safety research facilities. It is true that some of the major research projects such as LOFT, HDR and HSST operate under a number of bilateral agreements, but we see the need for more effective co-ordination between those different national projects that have similar objectives.

In this way the output from the different projects would become more easily comparable and thus more meaningful. It would also be possible to correlate and verify the results better. Ultimately this kind of co-ordination of projects would make the different researchers more familiar with each others' work. Such co-ordination could also help in the longer run to reduce the wasteful duplication which unfortunately affects 30 to 40 per cent of the safety research going on in the OECD countries. I should mention that some progress has in fact been made with regard to a number of projects in fuel cycle research. An international co-operative agreement covering certain national projects connected with the transport of spent fuel is going to be set up under the auspices of CSNI early next year.

Finally, CSNI has begun to prepare comprehensive state-of-the-art reports which are intended to review and evaluate research results in highly specific areas. As well as providing authoritative reviews, these reports will help in assessing progress in the enormous safety research programme now in hand. Altogether, the Committee has identified some 50 specific topics which should be assessed as rapidly as possible during the coming years. Last month we published the first report of this series. It deals with nuclear aerosols in reactor safety. On the basis of this experience we are now planning a round of consultations in capitals where we intend to more clearly explain our ideas about the need to find new and increased means to support this wide and expanding programme of international co-operation.

I would like to take this opportunity to discuss some of the objections we have encountered when discussing multilateral

co-operation in the framework of OECD. The most common reservation concerns the many bilateral research agreements that have been concluded, primarily between the United States and certain European countries and with Japan in a much more limited way between some of them excluding the US. The attraction to a country of working with another on a bilateral basis is understandable and has certainly its merits. However, something often overlooked is that nuclear safety concerns all of us and that to entirely rely on such preferred links is not only short-sighted but could prove fatal for nuclear power as a whole. We must remember that most of the 24 OECD Member Countries are not physically in a position to build up and maintain a system of bilateral contracts, as for example the United States. In this respect we often find that a small country may have such a special link with the US but not with some of its neighbouring countries. I would also point out the obvious fact that the level of penetration into many safety issues will necessarily differ between the US and some smaller nuclear countries; this disparity certainly does not add to the efficiency of these bilateral contacts.

Countries often also advance the argument that they need to be in total control of their own safety research to provide timely support for their licensing decisions. Clearly this rule is already compromised where bilateral and trilateral agreements on specific projects have been concluded. In any event this argument is not relevant to the medium and smaller countries which either conduct no safety research at all or operate projects on a very few specific topics.

To conclude, we have recently been told that IAEA should assume a bigger role in nuclear safety. In the OECD we have always insisted that the IAEA, which comprises virtually all the world's countries, should be the responsible organisation for codifying nuclear safety experience, which we see as one of the most important tasks in this field. It is also a natural role for IAEA to stress, at the political level, the importance of international co-operation in nuclear safety. To this extent we

welcome and support the Vienna initiative for a large international conference on nuclear safety next year. However, we believe strongly that the type of co-operation provided under the auspices of CSNI has no place in the context of IAEA. The reasons are obvious. Safety research as such does not exist as an independent discipline in the East block countries, or to say this differently, virtually all nuclear safety research is done in the OECD countries. There are also fundamental differences in safety provisions which would make, for example, international standard problem exercises on ECCS and containment impossible in the IAEA framework. There is also some doubt whether operational co-operation such as would be required for preparing coherent state-of-the-art reports, setting up data banks and interlinking safety projects is possible with eastern and with developing countries.

Gentlemen, I have taken a lot of your time to tell you about our plans and concerns. I have done so deliberately because I believe that the debate about nuclear safety and its international aspects is only just beginning.

Before making room for the experts I should like to offer the thanks of the OECD and CSNI to the Spanish Junta de Energia Nuclear and in particular to my friend Dr. Alonso for having invited us again to Madrid for this meeting. I have no doubt that in their efficient hands the meeting will be a success from the organisational point of view. It is now up to you, Ladies and Gentlemen, to make it rewarding from the professional point of view.

I wish you a successful meeting.

CSNI SPECIALIST MEETING ON REGULATORY REVIEW IN THE LICENSING PROCESS

Madrid, 7 - 9 Nov. 1979

REGULATORY REVIEW IN THE LICENSING PROCESS IN NEA MEMBER COUNTRIES

Pietro Giuliani

CNEN-Italy

Abstract

A review of the answers given by NEA Member States to a questionnaire prepared by a CSNI-Sub Committee on Licensing Working Group. The discussion of these answers in the W.G. was helpful in the preparation of this Specialist Meeting.

The CSNI Sub-Committee on Licensing, in its November 1977 meeting, proposed that a Working Group be established to evaluate the possibility of holding a Specialist Meeting in 1979 on the Regulatory Review in the Licensing Practice in OECD/NEA Member Countries. Consequently, in order to provide a basis for the preparation of the meeting, the Working Group issued an "ad hoc" questionnaire, which was sent to CSNI Member States.

The four main parts of the questionnaire dealt with the following aspects of the Regulatory Review:

- Technical bases for regulatory review
- Scope and stages of regulatory review
- Technical methods of review and assessment
- Position of regulatory bodies on special technical problems.

Answers were then received from 13 Member states, namely Austria, Belgium, Finland, France, Germany, Italy, Japan, the Netherlands, Spain, Sweden, Turkey, the United Kingdom and the United States. The questionnaire is attached to this report.

I. Technical bases for the Regulatory Review

Nuclear energy programmes have been established in all the countries which have answered the questionnaire. In all those countries but one (Turkey), construction and operation of nuclear power plants (NPPs) has been going on for a very long time already.

Basic nuclear energy legislation has been enacted in all of them as indicated in the answers and in a 1977 OECD/NEA publication on the licensing systems of NEA member countries, which has been used to supplement the information received.

In the area of safety criteria, standards and guides, the national positions, as described by the answers, show a good spectrum of cases(Table I) .

In some countries, as in the U.S., a very well developed set of basic safety criteria and standards exists; these criteria and standards have been evolved through the years, with a large effort involving many different organisations,national laboratories, agencies, professional associations and a large number of individuals. The material produced, well-known to everybody working in the nuclear field, has proven its worth and is continuously being updated and revised.

A few other countries have produced general safety criteria for NPPs; but, with the exception of the F.R. of Germany, have not yet developed a full set of guides. Guides and standards on a number of specific subjects, however, have been developed in several countries. Some countries, like France, are presently developing a set of general criteria and safety guides, of which a few only have been published officially so far.

Where a good background in conventional electrical industry had already been developed for some time, existing standards have been adapted from "conventional" areas to the nuclear energy field.

Some countries, like Belgium, the Netherlands, Spain and Turkey, adopt U.S. criteria and standards.

A few more, like Finland, have apparently based their own criteria on U.S. criteria.

Italy has adopted a few U.S. criteria and standards, with some modification to take into account the national situation.

Case by case review is used in countries where national criteria and standards are not developed and country of origin of NPP standards are used.

In all countries answering the questionnaire, work is going on vigorously in the development of guides and guidelines in the various fields of regulatory activity, with both procedural and technical aspects being taken into consideration. Site criteria and site evaluation guides have been developed in many states, with all aspects of site analysis and external events of different origin being taken into consideration.

A summary list of the material being prepared is given in Table II.

Work being done by International Organizations is followed everywhere with interest. France, the U.K., Turkey, among others, have specifically quoted documents and codes produced by the IAEA, the CEC, the ICRP and other agencies.

II. Scope and stages of regulatory review

The regulatory review appears to be conducted along basically very similar lines in the countries answering the questionnaire. The main stages of the review almost everywhere are the construction licence and the operating licence ones.

In some countries, these two basic stages are sub-divided into others, like in France, the F.R. of Germany, Finland and Spain.

Italy has a separate procedure for the approval of the site, which is reviewed separately from the plant.

In the U.S., the NRC adopted a rule on early site review intended to make siting decisions possible before there is a critical need for the site.

In Italy, again, the regulatory review is performed going through the approval of detailed designs, into which the plant is divided.

The regulatory review is conducted on two fundamental documents, a PSAR and a FSAR in the majority of the countries investigated. These documents are almost always based upon a format.

In many countries, this format is the one described in the U.S. Regulatory Guide 1.70, i.e. the U.S. NRC Format.

It is adopted by Austria, Belgium, Netherlands, Spain, Sweden, Turkey and, of course, the U.S.. Finland has based its format on the same document.

France, the F.R. of Germany, Italy, Japan, the U.K., have their own format; the Italian document, which refers only to the PSAR, is based mainly on the U.S. one, with a certain widening of the scope in the environmental area.

The U.K. format is agreed between the NII and the licensee.

The time-scheduling of the regulatory review appears to be rather independent of procedure, and relatively similar in the various countries: possible variations are due to conditions which may be different in different countries.

III. Technical methods of review and assessment

This question was sub-divided into four points, dealing with the methodology employed, the degree and type of independent analysis, the expertise required for the regulatory review and the independent specialist bodies utilized as support, and the sources of data.

Practically in all cases the methodology applied in the regulatory review and assessment is a deterministic one. In this type of approach, engineering judgement is applied against basic safety requirements. However, as the U.K. answer states, "both probability and reliability analyses and supporting data are expected to be a part of the licensee safety case and are used in the assessment whenever this is an appropriate technique". This statement could be applied to the situation of several other countries.

In many countries the U.S. Standard Review Plan is used as well as probabilistic approach in the area of external events.

Most Regulatory Bodies perform a certain amount of independent calculations in sensitive areas, using computer codes which may have been developed specifically for regulatory purposes. Sometimes these calculations and analyses are performed by outside consulting firms or by Universities.

All countries questioned have described their regulatory staffs as well-equipped to handle the work of safety analysis.

Several countries, such as the U.K.n the U.S., Italy and France, have senior advisory groups like the Advisory Committee on the Safety of Nuclear Installations, the Advisory Committee on Reactor Safeguards, the Technical Commission and

the Permanent Reactor Group.

In some countries, such as Belgium and Eire, the advice of international groups, such as the CEE, is called upon.

IV. Position of Regulatory Bodies on special technical problems

The fourth main section of the questionnaire dealt with a number of very interesting questions: Applications of the quantitative risk assessment, problems of reassessment and backfitting, the relationship between site and plant, occupational radiation exposure, decommissioning, and relationship between research and regulatory activity.

All of these points are extremely important in the licensing process and have been the object of much discussion in the last few years. The situation which emerges from the answers received is somehow similar for the various countries.

Let us try and examine the various items separately:

a) Quantitative risk assessment

The probabilistic analysis is, of course, one of the most important developments in the field of nuclear safety. Most countries with nuclear programmes have been studying its application to the licensing process. However, no country among the ones questioned makes full use of it in the regulatory review.

The U.S. answer says that the application of the methodology used in the Reactor Safety Study to the licensing practice may be impractical for a number of

reasons, such as the large volume of material that would have to be incorporated in the licensing application, the need for retraining a very large number of people, and the lack of reliable reliability data in some critical areas, etc.

But several countries felt that probabilistic analysis had considerable potential in conjuction with the deterministic analysis used traditionally. The U.K. answer, in particular, says that probabilistic analysis is considered as a necessary guide for judgement, even though a safety case cannot be based solely on this approach. When reliability data are available, the probabilistic approach is useful and sometimes essential in systems analysis.

In France it is felt that probabilistic methods will play a significant role in the licensing practices only if they are introduced progressively at various levels. In this respect, licensing authorities require that a reliability analysis for certain safety systems be included in the final safety analysis report for reactors under construction.

In many countries these methods are used in the comparison of different system solutions. This is done specifically in Sweden, Italy. The F.R. of Germany, Finland, Austria, the Netherlands, use probabilistic analysis in the assessment of safety of specific systems.

The picture emerging from the answers received is basically this:

Probabilistic analysis is slowly making headway in the licensing process, is being used for system comparisons and for specific analyses; its acceptance is increasing, but more slowly for the determination of levels of risks.

b) Safety re-assessment and back-fitting

In this area, few countries have developed formal criteria. In the U.S., the NRC may require the backfitting of a nuclear power plant and such action provides substantial additional protection.

In Belgium and Italy a safety reassessment of NPPs is required every ten years.

In Sweden, an appropriate policy is being developed in the area.

However, most countries have performed reassessment and backfitting on their plants, in particular countries where NPPs have been in operation for many years.

Areas which have been involved in backfitting are the emergency cooling systems, the emergency power supply systems, the radwaste systems, among others.

From the answers received, it is possible to predict that many interesting developments will arise in the future in this specific field.

c) Relationship between site and plant

All countries require design protection of NPPs against external events. Evaluation of design basis events is generally required.

Events with a certain probability of occurrence and with potentially significant radiological consequences are taken into account. Actual threshold frequency values per reactor/year appear not to be clearly defined.

Design basis events are determined for the site rather than for a particular reactor system.

Events taken into consideration are earthquakes, floods, extreme meteorological phenomena, airplane crashes, external explosions, large fires.

Several countries have developed or are developing criteria in this field. A few others, in their replies, have mentioned the growing body of safety guides being developed by the IAEA in the siting area.

d) Occupational radiation exposure

The ALARA (as low as reasonably achievable) principle has been generally applied in NPP built and operated up to now.

In all countries, the prevention of undue radiation exposure of personnel and public is achieved through sound design and appropriate operational limits and conditions. The goal is to limit individual and collective doses as far as possible.

In Sweden, a level for collective doses of 0.2 manrems per MW and year, is aimed at. A problem pointed out by Sweden, which deserves attention, is that sometimes the desire to keep radiation exposure to the minimum conflicts with the needs of nuclear safety. A good example is given by some types of inservice inspections.

Studies aiming at a better understanding of the occupational radiation exposure problems have been started in countries such as the Netherlands and Italy.

From the answers received it is clear that radiation protection problems are followed with great attention everywhere and that the most important activities for these considerations are maintenance and inspection.

e) Decommissioning provisions

The problem of decommissioning is tackled differently in different countries.

A few countries, like the U.S., the U.K. and the F.R. of Germany, have formal decommissioning provisions.

In the F.R. of Germany, the General Safety Criteria require them; in the U.K., the NII Safety Assessment Principle document includes a section on decommissioning provisions; in the U.S., NRC decommissioning requirements are set forth by 10 CFR. Furthermore, in the U.S., the NRC has an extensive decommissioning criteria developing programme.

In Spain, the Regulatory Body requires the applicant to submit decommissioning provisions at the construction permit stage of the regulatory review.

In other countries reviewed, there are no formal provisions or criteria in this area.

It is quite clear that this area will become more and more important in the future, both because of the number of nuclear facilities which are reaching "old age" and because of the growing concern over the environmental aspects of energy.

f) Relationship between research and regulatory review

In this field, there is a rather wide spectrum of answers. In the U.S. the Office of Nuclear Regulatory Research is one of the offices of the USNRC, so that the research programme is one of the activities of the Regulatory Body itself. Among the purposes of the research programme of the NRC is the verification of the adequacy of safety margins of the licensing requirements and the determination of technical bases for regulations and policies.

In some other countries, like Sweden, research is carried on by departments of the Regulatory Body itself; in the F.R. of Germany, there is a very close relationship between regulatory and research activities, even though they are not carried out by the same body.

In France, the Regulatory Body is supported from the technical standpoint by an external Institute located within the research and development "agency", so that nuclear safety research is co-ordinated with research and development activities in general.

In Italy, the Regulatory Body being a Directorate of the Atomic Energy Commission, there is a good tie between regulatory and research activities; other safety research is done by universities.

In the U.K., research work in the safety field is performed when necessary as support for assessment and review by the UKAEA, universities, national laboratories etc; all these agencies are not a part of NII.

In the other countries questioned it appears that not much nuclear safety research is being carried on.

g) Special problem areas

The U.S. answer mentions the NRC programme for the resolution of generic issues related to NPPs.

The U.K. mentioned a number of special problem areas such as application of probability and risk analysis, validation of codes, validation of reliability data, earthquake design and several others.

Austria mentioned ATWS, operational transients, emergency planning.

It may be interesting to report that the USNRC in its annual report to Congress in January 1979 planned

to present a list of 14 "unresolved" nuclear safety issues. Those issues, among which there are several important items, such as seismic design criteria, ATWS, PWR steam generator tube integrity, BWR nozzle cracking, are a good example of special problem areas.

V. Conclusions

The analysis of the answers to the questionnaire has shown several interesting things:

- The regulatory review in the licensing process and its methodology are very similar everywhere;
- Countries with power reactors of indigenous design have prepared more or less extensive sets of criteria, standards and guides, which tend to cover all aspects of nuclear safety, while countries which buy their reactors from foreign vendors tend to adopt country-of-origin criteria and standards. Sometimes these latter countries develop criteria and guides in the siting area and in the site-plant relationship area.
- All countries have active interest for the probabilistic analysis in the regulatory review.
- Several countries already use probabilistic analysis in conjunction with deterministic analysis or for system comparison.
- Decommissioning is beginning to be an issue in regulatory review even if not many countries have yet formal decommissioning provisions.

- Safety reassessment of operating NPPs is a formal requirement in some countries, but apparently (and logically), it has been going on practically everywhere. New safety research and more stringent safety requirements may cause the introduction of formal provisons in this area.

- Backfitting, a consequence of reassessment, has been going on in various areas of NPPs. It is a subject of great practical interest and will deserve a lot of attention in the future.

Furthermore, it was apparent that both national politics and public dissent are becoming more and more important factors in the licensing process.

On the basis of the discussion of the answers given by Member States, it was decided by the Working Group to organize the Specialist Meeting along the same lines of the questionnaire, adding a special session on the events at Three Mile Island and their impact on the regulatory process.

TABLE 1

Safety criteria, standards and guides

Austria	No Austrian safety criteria,standards and guides have been developed.In general,U.S. or FRG criteria and standards are used,after comparison with Austrian laws and regulations.
Belgium	No safety criteria,standards and guides of national origin are used. USNRC criteria and standards area adopted for the next four NPP's.
Finland	General Design Criteria for NPP's have been developed,on the basis of the U.S. ones. A set of guides on the procedural aspects of licensing is in preparation.
France	General technical regulations are being drafted. Use is made of foreign or internationally developed standards and guides, when applicable. Codes and standards are available on specific aspects. (Decision of 26 February 1974 on the main primary circuit of water reactors).
F.R.Germany	Safety Criteria for NPP's have been developed.An extensive set of Government Guidelines is available and is increasing. Guidelines of the TÜV.
Italy	Standards and criteria of the country of origin of the NPP are used, after comparison with Italan laws and regulations. A set of Technical Guides on procedural and technical aspects of regulatory activities is in issuance. Siting criteria have been developed.
Japan	A set of Guidelines is being issued by the AEC and is used in the regulatory review.
Netherlands	USNRC criteria and standards are adopted.National guidelines have been developed for external events.

Spain	No formal standards of national origin at the moment, but are being developed. Country of origin and/or international standards.
Sweden	U.S. General Design Criteria are used as a base. National standards available for certain technical areas of NPP's. A case by case review is performed.
Turkey	U.S. Criteria and Standards are used, together with International Standards and Guides. Regulatory and Safety Guides are being developed.
United Kingdom	Licensee is responsible for safety and develops its own design safety criteria. BSI-ASME-IAEA--ICRP standards are used, as well as CEE requirements for radiation protection.
U.S.	Basic Safety Criteria are the "General Design Criteria for NPP's". The National standards programme is carried out under the aegis of ANSI. The Regulatory Guide programme is in progress. The Standard Review Plans are used for guidance of staff performing safety review.

TABLE 2

Scope and Stages of Regulatory Review

Austria	Format: based on Reg.Guide 1.70 Preliminary Safety Report Final Safety Report - construction licence - operating licence
Belgium	Format: Reg.Guide 1.70 PSAR } one report FSAR } - construction licence - operating licence
Finland	Format:based on U.S.Reg.Guide 1.70 PSAR FSAR - construction licence - operating licence - fuel licence Safety evaluation report is issued by Regulatory body
France	Format: national origin Preliminary Safety Report Provisional Safety Report FSAR Decree authorising the construction after pre. SAR Authorisation of loading after pre.SAR Authorisation of operation after FSAR
F.R.of Germany	Format: national origin Safety Report Provisional site approval Provisional design approval Construction permit Operating licence

Italy	Format: national origin Site selection,based on a specific type of NPP PSAR - detailed projects approval - operating licence
Japan	Examination of the safety of the proposed plant - construction licence - operating licence
Netherlands	Format: Reg.Guide 1.70 PSAR - construction licence - operating licence
Spain	Format:Reg.Guide 1.70 PSAR FSAR Safety Analysis Report by Regulatory Body - preliminary licence - construction licence(PSAR,with indication of reference plant) - operating licence
Sweden	Format: Reg.Guide 1.70 PSAR FSAR - construction licence - operating licence
Turkey	Format: Reg.Guide 1.70 Site approval - construction permit Fuel loading permit Reactor start up permit Fuel op. permit - operating licence

U.K.	Format: agreed btw NII and licence Reference design report Preliminary safety report Pre-licensing or pre-construction s.r. Station Safety Report
U.S.	Format: Reg.Guide 1.70 PSAR FSAR - construction permit - operating licence

QUESTIONNAIRE ON REGULATORY REVIEW AND ASSESSMENT PRACTICES

1. Describe the technical bases adopted for regulatory review and assessment in the licensing process (of nuclear power plants).

   In particular:

   1.a. List and describe national requirements, safety criteria, standards and guides which are currently applied.

   1.b. In case no formal safety criteria, standards and guides have been developed, explain the alternative bases utilised for carrying out the regulatory review (e.g. type of case-by-case review, etc.).

   1.c. Describe the application of the previously mentioned standards and guides, or alternative technical bases, to the actual performance of regulatory review and assessment.

   1.d. Discuss envisaged further developments, if any, in the field of safety requirements, standards and guides.

2. Illustrate the scope and describe the stages of the regulatory review and assessment.

   In particular:

   2.a. Outline the stages and the time scheduling of the regulatory review along the licensing process.

   2.b. Describe the areas of review and the topics which are the object of assessment in each stage mentioned in 2.a.

   2.c. Describe the format (and how it is arrived at) and give an outline of the content of the documentation produced by the applicant and by the regulatory body in the matter of safety review and assessment.

   2.d. In case of absence of a formal format for the safety documentation, describe typical safety documents prepared by the applicant and the regulatory body for licensing purposes.

3. Describe the technical methods applied for performing the regulatory review and assessment.

   In particular:

   3.a. Describe the methodology applied to safety review and analysis (e.g. methods for site evaluation, accident analysis, prevention analysis, deterministic or probabilistic analysis, etc.).

3.b. Discuss the degree and type of independent calculations and analysis carried out by the regulatory body, including a brief description of mathematical models and computer codes applied.

3.c. Discuss the expertise required for the various parts of the regulatory review and assessment; describe the types of independent specialist bodies utilised, as a support, by the regulatory body (advisory committees, specialised institutions, etc.) and their degree of involvement in the regulatory review steps.

3.d. Outline the sources of data utilised for the regulatory review and assessment.

4. Discuss the position of the regulatory body in your country on a number of special technical problems encountered in regulatory review, and the solutions eventually adopted or envisaged to those problems.

In particular, the above-mentioned discussion should concern the following problems:

4.a. Relative merits and limits of applicability of quantitative risk assessment (probabilistic analysis) techniques to the regulatory review for licensing purposes.

4.b. Problems of reassessment and backfitting and associated criteria for application to actual licensing.

4.c. Assessment of the relationships between site and plant from the point of view of external events affecting plant safety (e.g., design basis events, their determination and consideration in the site review and design review stages).

4.d. Problems and trends of occupational radiation exposure (review of causes, analysis of critical plant systems and operations, analysis of possible improvements).

4.e. Decommissioning provisions.

4.f. Relationships of research with regulatory review (degree of support of research to regulatory review, mechanisms for feeding research results into the regulatory review process, etc.).

4.g. Other special problem areas being considered in view of the regulatory review and assessment.

SESSION II

ORGANISATIONAL MATTERS

Chairman: J.K. Pfaffelhuber

Scientific Secretary: P.B. Woods

"ORGANISATION AND EXPERIENCE OF THE REGULATORY REVIEW IN THE UNITED KINGDOM"

R Gausden and P B Woods
HM Nuclear Installations Inspectorate
Health and Safety Executive, London, United Kingdom

This paper outlines the regulatory requirements and stages of the review in the UK, describes the organisation, management and working procedures of the Inspectorate and discusses the effort and timescale which have been found necessary for a satisfactory safety review. A number of problem areas in the review situation have been identified from experience and these are discussed and possible solutions proposed. They include the importance of early review of generic aspects of systems which are being considered for licensing, the place of safety principles and of detailed engineering assessment in the review process, the independence of the regulatory group's activities in its study of major technical issues and the use which should be made of outside expertise.

"L'ORGANISATION ET LA PRATIQUE DE LA REVUE RÉGULATRICE AU ROYAUME-UNI"

R Gausden et P B Woods

Ce mémoire expose dans ses lignes générales les exigences et les étapes regulatrices de la revue au Royaume-Uni, il décrit l'organisation, la gestion, et les procédés du travail de l'Inspectorat et il discute l'effort et la durée du temps dont on a eu besoin pour faire une revue satisfaisante de sûreté. Un nombre de sujets problématiques, mises en question dans la revue ont été identifié de la pratique, sujets que l'on discute et auxquels on propose des solutions possibles. Parmi ceux-là sont l'importance dès les premiers temps d'une revue des aspects génériques des systèmes que l'on considere pour autorisation, la position des principes de sûreté et de l'évaluation détaillée de génie dans le procédé de la revue, l'indépendence des activites du groupe régulateur dans son étude des questions techniques principales et l'utilisation qu'en devrait faire de l'expertise d'en dehors de l'organisation.

1. INTRODUCTION

In the United Kingdom the safety of nuclear installations is regulated by the requirements of the Nuclear Installations Act of 1965 as incorporated into the Health and Safety at Work Act 1974. Under these Acts no site may be used for the purpose of installing or operating any commercial nuclear installation unless a licence has been granted to a corporate body by the Health and Safety Executive and is for the time being in force. HM Nuclear Installations Inspectorate is that part of the Health and Safety Executive responsible for administering this licensing function. As a part of this function, the Inspectorate carries out safety assessments of any proposed nuclear power installation before a licence is issued and also during the subsequent construction and commissioning phases leading to full power operation.

The Acts lay down only general requirements for the safety of nuclear power plants. Specific requirements are a matter for the Health and Safety Executive to formulate and apply. These may take different forms, for example, as conditions, binding in law, which may be attached to the site licence, as requirements for a safety case related to different stages of the licensing process or as guidance set out in the safety assessment principles for nuclear power reactors and associated plant which the NII have developed for their own internal use.

The Acts make the licensee or intending licensee such as the Central Electricity Generating Board or the South of Scotland Electricity Board responsible for the safety of their plant and hence it is the Boards that are responsible for putting forward the safety case in support of a licensing application.

The scope and timing of the regulatory review depends on a number of factors which may vary according to whether or not the design put forward is new to the UK as a system for licensing. On the one hand there may be a system which has been developed abroad and is new in its commercial application to the UK, whilst on the other hand there may be a design which is a straightforward extension of one that has already been licensed. The amount of information which is required, and the timescale for the review, could well be different in each case. In the case of a design new to the NII a great deal needs to be known and accepted before licensing, though this may essentially be related to safety issues generic to the type rather than specific to a particular design. For example, one purpose of the generic review might be to enable safety advice to be put to Government to enable a decision in principle to be made to proceed further with the design. Once the design has been accepted in principle then a more detailed assessment can be carried out of the particular design intended for licensing. In the case of an established design, already licensed, the NII are familiar with the design and safety issues and are more likely to accept that they are solved or are capable of solution. Hence the amount of information, particularly new information, required and the timescales for the pre-licensing review should be less than before.

The NII is at present concerned with four reactor systems, all at a different stage in development in relation to the nuclear power programme in the United Kingdom. The organisation of the Inspectorate to meet the demands placed upon it, the requirements of the review at each stage of development and the experience gained in carrying out these reviews are described in the paper.

2. ORGANISATION AND RESPONSIBILITIES OF THE NUCLEAR INSTALLATIONS INSPECTORATE

The Nuclear Installations Inspectorate (NII) was set up in 1959 when the first four commercial nuclear power stations of the Magnox type were under construction and when a few small research reactors were in operation. Since that time nine Magnox stations have gone into operation and the Advance Gas-cooled Reactor (AGR) programme has got well under way. Two of the AGR stations have now gone critical and the remaining three are at an advanced stage of construction. There has also been an increase in the number of research reactors.

Under the Atomic Energy Authority Act 1971 the main fuel manufacturing and processing and isotope separation plants, hitherto a part of the United Kingdom Atomic Energy Authority (UKAEA), were taken over by specially formed private companies and thus became subject to the regulatory control of the Nuclear Installations Acts as did the Magnox type reactors at Calder Hall and Chapelcross.

The main work of the NII is in the areas of site examination and evaluation and control of development around sites; safety assessment of nuclear plant designs and of their commissioning and operating procedures; and compliance assurance by means of site inspection during construction, commissioning and operation. The organisation of the Inspectorate has changed and its size has increased over the past 20 years to meet changing demands in these areas as the nuclear programme has developed. In particular additional responsibilities for the fuel processing and isotope separation plants referred to above led to a considerable increase in staff which now totals almost 90 inspectors. The 1974 Health and Safety Act has involved a further increase in the Inspectorate's responsibilities in that the UK Atomic Energy Authority and Government establishments such as those of the Ministry of Defence fall within its cope so far as the safety of their nuclear operations are concerned. In particular, the Inspectorate is now responsible under the Act for ensuring the safe operation of the UKAEA's reactors at Winfrith (SGHWR), Windscale (prototype AGR) and Dounreay (prototype fast reactor).

Figure 1 shows the organisation as it has been over the last few years.

Three branches of the NII are based in London and are responsible for the work on commercial nuclear power stations. These branches deal respectively with safety assessment of established reactor systems and licensing of reactors under construction; licensing and regulatory control of operating plants, together with siting and environmental matters; and review of future reactor systems. A fourth branch (Branch 3) located in Liverpool is responsible for licensing and operational safety of research reactors, fuel processing and reprocessing, isotope separation plants and waste management.

It will be seen that responsibility for carrying out the regulatory reviews is placed with Branches 1 and 4. Branch 1 has within it two Sections consisting of the Section Heads and some thirteen inspectors who are responsible for inspection and safety assessment of the AGRs which are currently being contructed and commissioned at Dungeness 'B', Hartlepool and Heysham (all twin-reactor stations) and also for the pre-licensing review of the Pre-Construction Safety Report for the new AGRs at Heysham 'B' and Torness. In addition, they give advice and carry out specialist assessments for the operating branch, Branch 2, in connection with the operating AGR stations at Hinkley Point 'B' and Hunterston 'B' and also, to a lesser extent, the operating Magnox stations.

Branch 4 has three Sections, consisting of the Section Heads and some sixteen inspectors, and is responsible for the safety study of fast reactors, the generic PWR study, formulation of safety assessment principles and guides, and a very small effort devoted to the safety of nuclear merchant ships.

It will be appreciated that in carrying out a safety assessment of a nuclear power plant a wide range of expertise is required. The staff of the NII has been built up over the years to reflect this requirement. All are scientists or engineers professionally qualified at honours degree or chartered engineer level, with suitable experience in an appropriate field and with a wide range of expertise e.g. in electrical and mechanical engineering, reactor physics, health physics and in more specialised areas such as pressure vessel technology, control and instrumentation, heat transfer, fluid flow, metallurgy and chemical engineering. However, the comparatively small size of the Inspectorate means that in the two Branches dealing with assessment work there are only two or three, and sometimes only one, individual specialising in each area.

This concentration of effort and expertise amongst only some thirty staff, and yet with responsibilities for several different reactor systems, leads to a number of problems. One is in management of the different projects against a timescale which is largely imposed from outside either by Ministers asking for advice or by licensees or intending licensees who have their own programmes to meet. Another is to obtain the depth of study which is necessary in some areas.

The first problem was solved in Branch 4 by adopting the technique of matrix management. From 1974-78, for example, the Branch had three systems under review: the fast reactor, the PWR and the commercial SGHWR. Each Head of Section was made Project Leader for one of these systems and yet employed not just his section but members of all three sections, so as to provide the breadth of expertise that was required, on the project. Effort allocated to each project was controlled by having each assessor account for his time spent in $\frac{1}{2}$ day units reported by means of weekly time sheets. As a matter of interest, this showed that little more than half of the total time available was spent on an assigned major project, the rest was taken up with such matters as advice to other Branches, management of extra-mural projects, attendance at conferences, training and leave.

The second problem was solved by obtaining specialist support from independent individuals or bodies outside the Inspectorate. In a number of cases this was done by encouraging the building up of centres of expertise in the universities. The part played by these external bodies is described in greater detail later in this paper and in the two UK papers in Session VII. However, in general terms support obtained from outside bodies takes two forms. First is the extra-mural research work undertaken by universities and by external consultants from the specialist institutions such as the Welding Institute, British Hydrodynamics Research Association, Lloyds etc, which provide the NII with an independent source of research and advice on the various topics of concern. Second are those independent experts used by the Inspectorate as members of study groups convened as necessary to discuss and advise them on particular aspects of the safety of the system under review such as the integrity of pressure vessels, loss of coolant accidents and fuel behaviour. The extra-mural work may be continued over a number of years whilst it is relevant to the problems which are being assessed by the NII and the study groups generally provide their advice in the course of one or two years study in the form of a report or reports giving their opinion to the NII on the specific items they have been asked to look at.

In addition to those groups which advise the NII directly, there is also a senior independent body, the Advisory Committee on the Safety of Nuclear Installations, which advises the Health and Safety Commission, and Ministers, on major issues affecting the safety of nuclear installations including design, siting, operation and maintenance which are referred to it or which it considers requires attention. This Committee convenes working groups as necessary to look into specific questions in greater detail than can the main Committee. At present, for example, there are working groups on fast reactor safety and radioactive waste. Of course, these working groups report to the main Committee rather than to the Nuclear Inspectorate but the NII has access to their reports. The Advisory Committee's function is to provide advice on policy matters rather than become involved in the regulatory review as such but it does have a part to play in monitoring the work and advice provided by the Nuclear Inspectorate.

3. BACKGROUND TO THE REGULATORY REVIEW

The power to attach enforceable conditions to a site licence and to vary them at any time or to withdraw the licence itself provides a very flexible regime of statutory control in the UK. Yet it should be noted that in the period prior to issue of a licence the Executive and hence the NII has no formal powers over what is proposed for any site. However in practice a licence would not be granted unless sufficient assurances were obtained on the suitability of the site itself and on the proposed installations. Applicants wishing to construct a nuclear installation consult the Inspectorate on the information required and discussions are always held before formal procedures begin. The NII's assessment of the site and the siting policy which is followed have been fully described elsewhere [refs 1 and 2] and will not be discussed here. The nature and extent of the NII's assessment of the design of an installation before issue of a licence depends on whether the plant is of an established or a new type. In the case of the Magnox and AGR nuclear power stations, the design of which have followed a pattern of steady development in the UK and which are owned and operated by only two licensees, it may be possible to avoid extensive pre-licence examination or to specify the Inspectorate's information requirements in detail. In general however these requirements are similar to those given in IAEA Safety Series No 34 [ref 3]. For reactor types on which no operational experience in the UK is available or where, as in the case of the fast reactor, the system is still at the prototype development stage, pre-licence negotiations and information requirements are necessarily very much more extensive and protracted.

When the Inspectorate is satisfied that the proposed installations should be capable of being engineered to meet its requirements for licensing, the applicant is advised of this and he may then make formal application to the appropriate Secretary of State. On receiving the application, which will be for the construction and operation of a particular reactor system on a specified site, the Secretary of State will normally direct the applicant to publicise the proposal and give notice to specified Public and Local Authorities including River Boards who have the right to make representation regarding the proposal within 3 months. The procedure to be followed is laid down by the Electric Lighting Act 1909 under which the Secretary of State's Consent is required before a power station of any type can be built or extended. When all interested parties have been given an opportunity to comment or object to the proposed station, the Secretary of State decides whether or not the proposals affect their interests to an extent which makes it desirable to hold a public inquiry. If, however, the Local Planning Authority objects, the Secretary of State is obliged to hold such an inquiry. A nuclear site licence will not be granted until the Secretary of State gives his Consent to build a nuclear station under the Electric Lighting Act.

Having obtained Consent for use of the site, the applicant then requires a licence and for this to be granted a satisfactory safety case must have been provided for review by the Inspectorate.

4. REQUIREMENTS FOR LICENSING

At the prototype stage in a design which is being developed for eventual inclusion in the nuclear power programme, or where a reactor system has been so developed elsewhere but is new to the UK, the safety review carried out by the NII is less formalised than would be the case for a system which was well established in the UK. The aims in such a situation are to build up sufficient expertise to be able to independently assess the safety of the new system, to explore and become familiar with the main safety issues in relation to a generic design and to advise the nuclear industry on aspects of the design or the safety argument being put forward where these might prejudice future acceptance for licensing. Hence a flexible approach is required, dependent upon the system being reviewed and the stage of development which has been reached.

The stages of the regulatory review and the areas to be covered once a design has been accepted in principle for licensing generally follow the timetable of safety submissions required from the intending licensee. The basic elements have been developed over the years but are now essentially as follows:-

(i) Design safety principles and criteria. These set down the criteria and requirements used to provide for an acceptable standard of nuclear safety and hence radiological protection to the public and to the operating staff during normal and abnormal operating conditions. They provide guidelines for the type, standard and performance of the protection equipment which is provided in the design to meet the radiological principles, and embody design safety criteria to demonstrate that the completed plant will meet the safety standards required by the NII;

(ii) Preliminary Safety Report (PSR) - Stage A Submission. This is prepared on completion of the main features of the reference design and gives a preliminary description with outline drawings including layout, main and auxiliary plant, electrical and control and instrumentation systems together with a comprehensive set of reactor parameters. It shows the principles by which the reference design can meet the design safety criteria and provides a preliminary safety analysis of the critical fault conditions and preliminary assessment of the performance and standard of the proposed protection equipment;

(iii) Safety case and fault studies: information provided in the PSR has to be supplemented by safety cases and fault studies covering the main safety issues;

(iv) Detailed research and development and component development programmes are required including information concerning timescale and objectives;

(v) Quality assurance and in-service inspection proposals including the intending licensee's QA programme and management scheme and the QA and QC schemes for the main contractor and for their sub-contractors, for the main safety related items;

(vi) Contract design. Details of all safety related items i.e. the specific design intended for licensing;

(vii) Pre-construction Safety Report (PCSR) - Stage B Submission. This consists of a description of the contract design and a more extensive safety assessment of the design of the nuclear station than Stage A. It will include a detailed safety case covering all foreseeable fault conditions for which protection is provided and for those parts of the plant where a very high integrity is claimed.

The timing of these stages will vary from case to case. For a system new to commercial operation in the UK (i) and (ii) would be required about 2 years before the date of licensing; (iii) and (iv) about 12-15 months before licensing; and (v) and (vi) about 6 months before licensing. For a system already licensed the period of review could be much shorter than this.

The information provided by the time the NII are required to give advice to the Executive on licensing should be such that detailed design, manufacture and construction can proceed with small risk of significant modifications subsequently being required for safety reasons. The attainment of this objective, seen as highly desirable so that both the NII and the intending licensee are confident that the design intent and safety principles will produce an acceptable reactor installation, may require, especially in the case of a system not previously licensed, that stage (vii), the PCSR, has been received and examined.

It is worth mentioning that at about the time the Contract Design details are received, if not before, any queries relating to siting should have been cleared e.g. with regard to external hazards and control of housing and other development around the site.

The topics covered by the PSR and the PCSR are essentially the same in each case, but with more detail and a greater number of supporting documents in the PCSR as is appropriate for this later stage. A typical list of the contents of these submissions is given in Appendix I.

It is fundamental to the regulatory review and assessment that it should be based on the design information, safety case and supporting data provided by the intending licensee as part of their safety submission, and they are expected to make this as comprehensive and searching as might be required. Sources of technical information will include the UKAEA (who provide supporting R & D for all reactor systems developed in the UK) the Systems Reliability Service (for reliability data) and the designers (vendors) own development programmes. The intending licensees, especially the CEGB, also have their own research laboratories which they will use as a source of data. In addition, however, the NII will obtain independent data from its own extra-mural contracts, consultants and study groups should this be thought necessary. Both the licensees and the NII make use of data obtained worldwide where this is relevant.

It has to be demonstrated in the intending licensee's safety submissions that the proposed designs meet their criteria, and these submissions are in turn assessed by the Inspectorate whose staff apply the NII's safety assessment principles [ref 4] in their review. It is a requirement of the NII's principles, for example, that international and national recommendations on health and safety are followed and that the design is engineered to a sufficiently high safety standard which will satisfy the relevant codes.

5. ASSESSMENT FOLLOWING LICENSING

Once a licence has been granted, the licensee may start construction of the plant. The NII maintains regulatory control over construction, commissioning and subsequent operation by means of conditions attached to the licence. The various stages of this work, related to site construction and commissioning,

are described in a paper presented in Session IV. At the same time, assessment of the licensee's safety case continues in detail and further stages of the safety case are submitted.

The Stage B submission will be subject to amendment and further review as contruction proceeds to cater for any design modifications which are subsequently required. More importantly, it is at this stage that the NII carries out a comprehensive assessment of the engineering details of the design to ensure that the construction, commissioning and operation of the plant meets the agreed design intent.

As assessment proceeds on the detailed design, and results of longer term experimental work and final development tests become available, and as defined sections of the plant reach completion the licensee is required to present safety reports for each section in turn. In due course these will be included with an updated and more detailed fault analysis in a single comprehensive document, the Station Safety Report or Stage C Submission, which will form an authoritative report by the licensee to be used throughout the life of the plant and will form the basis for the NII's own acceptance of the station. When approval is required to raise power this safety report will have been updated as necessary following commissioning tests and further lengthy experimental programmes and will therefore include more specific information on certain safety topics than was available previously. More information may also be required on matters relating to plant operation. Hence the Stage C Safety Report should be a complete and comprehensive document by the time the station reaches commercial power operation.

As has been explained, the timing of these stages will vary from case to case. Whilst the pre-licensing review and assessment could take between one year for a design of a type already licensed to perhaps two or three years for a novel design and is a matter for the NII to decide, the post-licensing stages will of course depend on progress with the design and its construction and hence the timing is very much subject to the programme followed by the project overall. Similarly, whilst the general form of the safety case and its review procedure is as described it is not laid down as a statutory requirement but may be varied by the NII. Thus a flexible approach is possible and changes may be readily introduced if considered necessary.

Whilst the above review is proceeding, the NII also have to report on its activities. This is done by means of internal safety assessment reports which set down its view at each stage of a safety case put to it so as to provide a record of its work and to support the advice it gives to the Executive and to Ministers. These documents may include: a pre-licensing generic safety review, a pre-licensing assessment of a specific design proposal (perhaps based on the Stage A and part of the Stage B Report), a preliminary assessment of the design details prior to fuel loading based on the Stage B report and a final assessment of the design safety report (Stages B and C) either prior to raising to power or prior to commercial operation at power as appropriate. The NII's reports are not normally available outside the HSE although in some cases a non-classified open executive report is written summarising their findings.

6. EXPERIENCE OF THE REVIEW

6.1 Prototype Development

In the UK the fast reactor has been developed by the UKAEA through several stages, from the original Dounreay FR and the Prototype FR now operating at

Dounreay to the present design study for a Commercial Demonstration Fast Reactor. The NII have kept a watching brief on these early designs but since 1974 greater effort has been applied in carrying out a review of the generic issues of fast reactors, with particular attention being paid to the design concept being developed in the UK.

The main objectives during this early review period have been:-

(i) to build up a sufficient level of expertise to independently assess the safety case put forward with a licence application;

(ii) to advise Ministers, the HSE and the Health and Safety Commission on fast reactor safety; and

(iii) to advise the nuclear industry on aspects of the design or the safety arguments being put forward, in particular where it was felt that proposals were likely to lead to difficulties at the licensing stage.

To achieve these objectives a nucleus of expertise has been built up within the NII managed by a Head of Section. The work has concentrated on safety issues which were seen to be fundamental to the fast reactor concept since a proper scientific and technical understanding of these issues would be necessary in any subsequent assessment, whatever the detailed design. These issues were, for example, the whole core accident and the protection and style of containment which could be provided against such an accident, sub-assembly faults as a possible trigger for whole core events, and molten fuel/coolant interactions.

The approach has been to encourage members of the team, and only a small number of assessors was available, to develop their knowledge of the issues in some depth through their own study, by contact with industry, the UKAEA and research bodies, by means of extra-mural contracts placed by the NII and through selective attendance at conferences both national and international. Contact with those in the field is also maintained through committee work and by means of information exchange agreements with other regulatory bodies.

A problem has been to maintain a sufficient level of expertise over the years as the possibility of an early commercial fast reactor in the UK has receded. The matrix management system described earlier has been found to provide a solution since it allows a larger number of assessors to familiarise themselves with the fast reactor system without taking an undue proportion of the total effort available.

In parallel with this activity of paying attention to the main effects of an accident in the fast reactor, and its protection, some five years ago the NII formed a group in the future systems Branch which set out to examine the system in detail, starting from a fault schedule and the sources of radioactivity present, and using the techniques of fault and event tree analysis. This examination was intended to explore the protection provided against faults or failures associated with the generic plant and hence reveal any unknown fault sequences not effectively protected against and hence not so far explored in relation to the proposed design and its safety case. This group has not yet completed its work but has already pointed to the need to pay more attention to certain key structural features where in-service inspection throughout life may not be practicable.

To maintain its interface with industry the NII is represented on various fast reactor committees within the UK, the most senior of which is the Joint Committee on Fast Reactor Safety. This Committee consists of representatives from the electricity utilities, the design company and the UKAEA and it has,

within its terms of reference, a requirement to formulate fast reactor safety criteria as a basis for future designs. The opportunity is available, therefore, through this Committee and through its other contacts with the nuclear industry for the NII to convey its views, in particular on matters which are believed likely to cause difficulty in licensing. Our role on these committees is advisory only, and the industry is under no obligation to act upon the NII's advice. Nor, in giving it, do the NII commit themselves in any way. Nevertheless the advice must be sound and is expected to reflect the view which would be taken in carrying out the review at the licensing stage. There are difficulties in this, because developments on the research side leading to a better understanding of the problems or revealing further problems could change the picture as could developments in safety philosophy. Also, advice already given could lead an assessor to prefer one solution rather than another. However, this approach is seen as the best way of working at this stage of the review so as to achieve the overall objective of ensuring an acceptable design for licensing.

The fast reactor review has involved the NII in some 15 man-years of effort, mainly spread over the years 1974-date and has incurred a further expenditure of some £800,000 by HSE in support work such as consultancies and extra-mural research. The CDFR is likely to be the subject of a public inquiry in due course and only then will licensing be considered. It is envisaged that the present commitment will continue for the future until a pre-licensing review of an actual commercial design is entered into, when an increased scale of effort will be required.

6.2 Systems New to the UK

In 1973 the CEGB declared an interest in a PWR, later confirmed as a Westinghouse design, for the next stage of their nuclear programme. This system would be new to the UK in terms of commercial application and until that time the NII had only kept a watching brief on safety aspects of LWRs in general.

In 1974 the Government chose the SGWR for the future power programme but required that the NII complete the review which it had started on the generic safety issues of the LWR. In practice this meant a generic review of the Westinghouse design of PWR. This review was completed in 1978 and a summary report of the NII's conclusion has been issued [ref 5]. Work in the NII is now continuing in preparation for a formal safety review leading to licensing together with a possible public inquiry.

For the purpose of the review the generic aspects of PWR were taken to be those safety issues which can be regarded as specific to and inherent in the concept and those features which, while common to other nuclear power systems, have novel safety significance in the PWR.

From the watching brief which had been kept on worldwide experience with LWRs and on the basis of problem areas inherent in all reactor systems the generic issues selected for detailed consideration were:-

(i) Potential plant faults and their analyses, including in particular the loss of coolant accident analysis.

(ii) Integrity of the primary coolant circuit, including reactor pressure vessel, primary loop pipework and the steam generator.

(iii) Fuel element behaviour.

(iv) Reactor protection system.

(v) Containment.

(vi) Radiological risk in normal operation.

(vii) Radioactive waste arising on the reactor site.

The objective of this study was to arrive at a view on the safety of the PWR concept and in particular to determine the technical conditions which would need to be satisfied for a PWR to be acceptable in principle for use as a commercial nuclear power plant in the United Kingdom. While a particular plant, the Westinghouse 4 loop 1300 MWe plant as at Trojan, Oregon, was selected as a reference and much of the detailed study was based on it, the review was not intended to be, and could not be regarded as, a commentary on a specific design or a particular plant.

To supplement this study the Kraftwerk Union PWR of similar size was also considered and discussions held with the Company. The same generic issues were examined sufficiently to enable a sound appreciation of the important differences between the two designs to be made. In addition, discussions were held with regulatory groups in France, Germany and the USA and information was obtained from relevant world and UK engineering and scientific studies including the Rasmussen Report [ref 6] and the UKAEA investigation into pressure vessel integrity carried out under the chairmanship of Dr W Marshall [ref 7].

It was clear at an early stage that the NII staff would need technical support in considering at least some of the more important generic safety issues which had been identified. Hence a substantial programme of extra-mural work was put in hand covering both theoretical and experimental studies in these areas. In addition two study groups were set up early in 1974 consisting of independent experts from the universities, research institutions and industry, one to advise the Chief Inspector on technical and scientific issues associated with the integrity of pressure vessel parts, the other to advise on LOCA/ECCS processes. A further, smaller, working group consisting of representatives from the Inspectorate, the UKAEA and the Authority's Safety and Reliability Directorate was formed to prepare a fault schedule and extend the fault analysis of this system to identify any so far unrevealed faults or faults against which the protection provided in the generic design was inadequate. Unfortunately from the point of view of the PWR study the first two of these groups had to turn their attention to SGHWR issues when, later in the year, this system was chosen for the UK nuclear power programme. They proved their usefulness in that role, whilst individual members continued to assist with the PWR review outside of the groups' activities. This switch of UK effort to the SGHWR also meant that the NII had to fall back on its own resources for the fault studies work.

The review procedure has thus been based on a critical examination of the generic issues of PWR. Taking account of (i) a study of the safety cases put forward by W and KWU, (ii) information gained from the various supporting activities and (iii) the background knowledge acquired from a continuing study of PWRs over several years, questions were framed and addressed to both these design organisations. Responses to these questions, coupled with information obtained from the original technical material and generic background material available to NII formed the basis for the conclusions and recommendations arising out of this study.

The scale of the review and the tight timescale within which it had to be carried out, especially when viewed in relation to the limited resources the NII had available for this work, gave rise to a number of problems, the solution of which has provided us with much worthwhile experience. Problems which arose included:-

(1) The need to build up a team of assessors with the right experience in safety assessment and in water reactor systems so as to be able to come to an independent view. Some recruitment was necessary and in the early stages a high proportion of time was spent on monitoring and reviewing experience from operating PWRs, literature surveys, visits to plants, involvement with EEC and OECD safety working groups (where the interest was predominantly LWR issues), study of fundamental scientific data and information e.g. related to fracture mechanics, heat transfer and two-phase flow, internal discussion groups to generate ideas and share information and the development of safety assessment principles and criteria;

(2) A significant difference in this review from similar exercises being pursued on other UK designed systems arose from the fact that in an off-the-shelf PWR the major design concepts are effectively frozen, whilst the NII is involved and can influence safety design in the UK systems as they are developed;

(3) There are items of plant or components, which although common to other reactor types, become wholly or partially specific in the context of the PWR by virtue of unique modes which may arise, e.g. the containment and certain aspects of the protection systems. This required a detailed understanding of the characteristics of the system under review;

(4) The original plant will have been designed against certain criteria or standards. These may be set at different levels from those normally accepted by the NII, or they may be aimed at the specific system being considered. This required assessment of each important case against the NII's own safety assessment principles;

(5) It was essential that the physical behaviour of the plant both in normal and abnormal situations was adequately understood. It had to be possible to express this behaviour in recognised and acceptable scientific or engineering terms. In this context, the plant was defined to include the fuel, primary coolant, emergency coolant and any other possible contributor to abnormal behaviour existing as part of the design. This aspect was particularly difficult for the NII since previous experience was with gas-cooled reactors with a single phase coolant and very different core transient characteristics. To achieve this understanding considerable scientific and engineering investigational work was initiated, for example, in single and two-phase coolant behaviour, circuit materials behaviour, and reliability studies of components and systems, as already described.

In all, the assessment work has involved the NII in a total effort of about 24 man-years and has involved a total expenditure to date of some £600,000 by the HSE on consultancies and associated work.

The general conclusion of the generic review was that there was no fundamental reason for regarding safety as an obstacle to the selection of a PWR for commercial generation in the UK. However, a final decision as to acceptability for licensing cannot be given on the basis of a generic study alone. A more detailed examination of an actual design intended for licensing is now necessary and this will follow the procedures outlined in Section 4 above.

6.3 Established System

A further system which the Inspectorate is concerned with and which is at present being considered for licensing is the Advanced Gas-cooled Reactor

(AGR). This system has been developed within the UK following the success of the Magnox Gas-cooled Reactor stations and taking the prototype AGR at Windscale, which went to power in 1963, as a model. The first two twin-AGR stations at Hinkley Point 'B' and Hunterston 'B' were licensed in 1965 and 1966 and went to power in 1977 and 1978 respectively. Further stations were licensed at Dungeness 'B' in 1965 and the similar twin-AGR stations of Heysham and Hartlepool in 1968 and 1969, and these stations are still commissioning. It should be noted that whilst these stations are all AGRs, and hence with the main features of this system in common, there are differences between them. Many of these differences are of importance to safety.

The current programme consists of two twin-AGR stations to be built at Heysham, as an extension of the present site, and at Torness. They are to be replicas of Hinkley Point 'B' in many features but, again, there are important changes which have been made to reflect the development of safety philosophy and requirements since this station was first commissioned.

Hence the Inspectorate has to carry out a pre-licence review so as to be satisfied with the design intent of the new AGRs before a licence can be granted. The stages of this review are as set out in Section 4 above. The PSR was received in July 1978 and was reviewed over a period of some 10 months employing the 10 assessors and 2 Heads of Section available for this work in Branch 1.

The NII's assessment of the PSR was based on its knowledge and experience of the safety standards which had been required for, and met by, the earlier AGRs and on the guidance provided by the Safety Assessment Principles [ref 4]. From the examination of the PSR which was carried out a number of safety issues and points which require clarification have been identified and have been discussed with the Generating Boards (CEGB and SSEB) and representatives from the designers, the Nuclear Power Company (NPC). As a result a good many points have been resolved and a more satisfactory position on many of the remainder should have been reached by the time the PCSR is submitted to the Inspectorate. The PCSR should therefore reflect the agreements reached on the outstanding safety issues and if this is the case then, following the agreed timescales, it should be possible to grant a licence some 21 weeks after receipt of the PCSR and its supporting documents.

There are particular problems in licensing further reactors of a kind already licensed. They are mainly concerned with the case for requiring safety improvements in the next stage of a system which has already been accepted without them and the question that is then raised as to whether or not to back fit. Regarding the first, these are argued against the currently desirable safety standards as set out in the Inspectorate's safety assessment principles, it being our view that it is practicable to do more from the start of a new design and still maintain the advantages of replication. As for back fitting, this is not normally required unless the matter is judged to be of appreciable safety importance: in carrying out its review the Inspectorate would compare the established system with present requirements and would then decide each case on its merits using the principle of requiring the licensee to make such changes as are reasonably practicable.

The pre-licensing review of the new AGRs is estimated to require some 12 man-years of Inspectorate effort, supported by some £100,000 of extra-mural support and advice from consultants.

7. FUTURE DEVELOPMENTS

An area of particular interest in recent years is the development of probabilistic analysis as applied to reactor safety. The NII is of the view that this is a necessary guide for judgement and a valuable discipline in the safety assessment process. The design safety criteria which have been put forward by the Generating Boards for the new AGR stations makes use of this approach and this has been accepted by the NII as a part of the safety case. It is also the intention, as will be seen from Appendix I, that a station risk assessment will be presented. This is a new feature of the review and will be looked at with interest. However, the NII does not consider that a safety case can be based solely on this approach. The state of knowledge of reactor systems, particularly the very important structural and mechanical aspects, is not yet such that this type of analysis can replace the deterministic methods and the application of engineering criteria which form the basis of our regulatory review.

We are also now requiring decommissioning to be considered in the safety case from the earliest stage. At least a statement of intent should be provided and eventual decommissioning acknowledged e.g. by providing records of materials used and drawings of the plant for this special purpose if for no other.

A re-assessment of the plant and its safety case following a year or two's commercial operation has also been carried out by the Inspectorate for the first two AGR stations. This provides a further check on late plant modifications and early operating experience and has been found to be well worthwhile. It is likely to be a requirement of the review process for all future stations.

Finally, we are now firmly of the view that the regulatory safety review has to be a continuing feature of the life of a nuclear power station. Operationally we have biennial shutdowns of one reactor (of each twin-reactor station) and this reactor cannot be started up again until a review of experience from the maintenance schedule has been carried out. As a reactor ages so it becomes necessary to include more detail in this review and also to consider a long-term review, extrapolating to a further 5 or 10 years operation to be further satisfied that safety problems can be anticipated and corrected without coming up against operational or economic pressures.

8. CONCLUDING REMARKS

During the last six or seven years the NII has had to develop the expertise and ability to carry out safety reviews for regulatory purposes of three different reactor systems (four if the SGHWR is included) each presenting different problems not just because of their different characteristics but also because of their different stages of commercial development in the UK. We are reasonably satisfied with the form of review which is now required and it is hoped that this account of our experience and requirements will be of value to others with similar responsibilities.

9. REFERENCES

(1) IAEA/SM - 169/23 "Licensing and regulatory control of thermal power reactors in the United Kingdom" by W S Gronow and R Gausden, 1973.

(2) "Application of safety and siting policy to nuclear plants in the United Kingdom" by W S Gronow. Proc Symp Env Contam, IAEA, Vienna, 1969.

(3) IAEA Safety Series No 34 "Guidelines for the layout and contents of safety reports for stationary nuclear power plants", 1970.

(4) "Safety assessment principles for nuclear power reactors", HMNII/HSE, HMSO, April 1979.

(5) "The Generic Safety Issues of Pressurised Water Reactors", NII 1979.

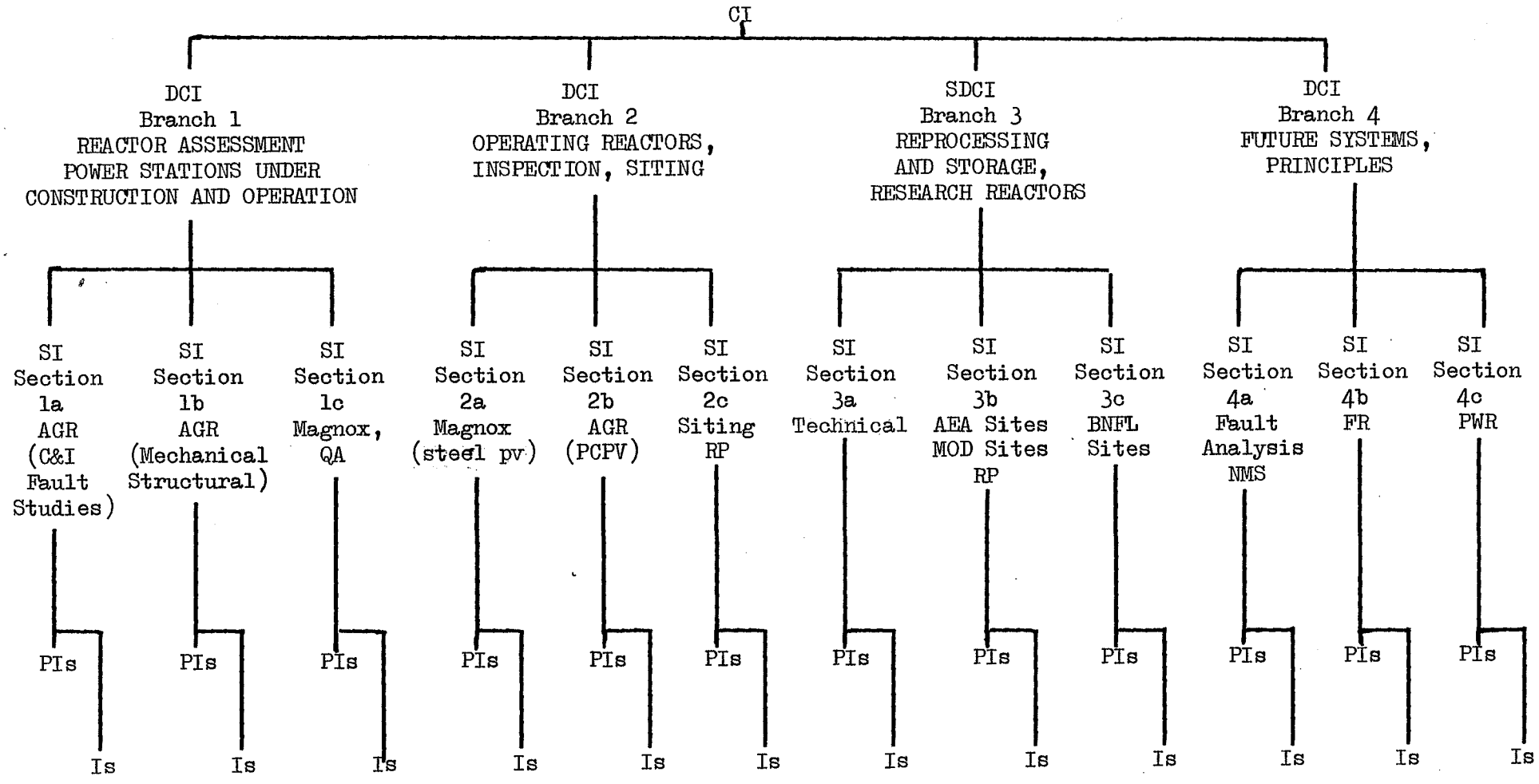
(6) "Reactor Safety Study. An assessment of accident risk in US commercial nuclear power plants" WASH 1400 (NUREG 75/014).

(7) "An Assessment of the Integrity of PWR Pressure Vessels - a report by a study group under the Chairmanship of Dr W Marshall", UKAEA Report LWR(77)1, 1977.

October 1979

FIGURE 1

NII STAFF TREE



APPENDIX I: TYPICAL CONTENTS OF SAFETY REPORT

Section 1 - Introduction: includes the objectives, the arrangement and layout of the safety report, the location of the site and a general description of the reactor.

Section 2 - Safety Principles: includes the design safety criteria and the safety principles and an outline of the safety case.

Section 3 - Site Description: may include meteorology, geology, ecology, district radiological monitoring programme, design requirements determined by the environment, and civil engineering design.

Section 4 - The Reactor: includes a description of the reactor core its nuclear characteristics, thermal and hydraulic characteristics.

Section 5 - Fuel: mechanical design and criteria.

Section 6 - Radiological Protection and Shielding.

Section 7 - Fuel Handling and Storage.

Section 8 - Reactivity Control Systems.

Section 9 - Reactor Cooling System: includes pressure circuit design.

Section 10 - Engineered Safety Systems.

Section 11 - Auxiliary Systems and Services: includes the reactor coolant circuit de-contamination systems, fire protection, instrument and general service systems, communication systems.

Section 12 - Containment.

Section 13 - Control and Instrumentation: includes the main control room and emergency centre, safety systems, direct alarm system, on-line computer system.

Section 14 - Radioactive Waste Management.

Section 15 - Station Layout.

Section 16 - Post Trip Operation.

Section 17 - Electrical Systems.

Section 18 - Steam and Feed Systems including turbine hall systems, main feed system, water chemistry.

Section 19 - Main Turbine Generators.

Section 20 - Initiating Faults.

Section 21 - Analysis of Faults: includes internal and external hazards.

Section 22 - Station Risk Assessment.

Section 23 - Safety Related R & D Programme.

Section 24 - Safety Related Technical Work Programme.

Section 25 - Quality Assurance.

Section 26 - Operations Intent.

ORGANISATION AND PRACTICES ON REGULATORY REVIEW IN THE LICENSING PROCESS OF NUCLEAR POWER PLANTS IN SPAIN

P. Trueba
Junta de Energía Nuclear
Madrid-Spain

This paper presents the actual organisation, practices and experience of the JEN Nuclear Safety Department on the regulatory review in the licensing process of nuclear power plants in Spain. Topies to be covered are: The structure, organisation, staff and principal functional areas of the NSD, the academic qualifications and work experience of the NSD personnel, recruiting and training, the conduct of the regulatory review during the licensing process and working procedures, the manpower and coverage of the different technical areas, the principal problems and conclusions.

1. INTRODUCTION

This paper presents the practice and experience of the Nuclear Safety Department (NSD) of the Spanish Nuclear Energy Board (JEN) on the subjet of regulatory review in the licensing process of nuclear power plants.

The licensing process of nuclear power plants in Spain is based on Nuclear Energy Law 25/1964 of 29th April 1964, further developed by the Nuclear and Radioactive Facilities Regulation (RINR) enacted by Decree 2869/1972. According to its provisions, the Ministry of Industry and Energy (MIE) is the national authority for licensing NPP through the Director General for Energy and after - statutory advice from JEN. Thereby the MIE is responsible for granting the governmental authorisations concerning siting, construction and commissioning of nuclear power plants, without prejudice of other approvals within the autohority scope of other Ministries and Administrative Agencies or of the Municipalities involved.

The formalities and procedures of the current Spanish licensing process have been sufficiently described in the document "Licensing Systems and Inspection of Nuclear Installations in NEA Member Countries" NEA, OECD (1977). But it is to be noted, nevertheless, that the MIE's Directorate General for Energy is particularly competent on:

a) Energy planning and coordination and, working together with JEN, preparation of programmes to bring nuclear energy into the national supply system. (Promotion activities).

b) Processing and granting of governmental licenses as required by law after hearing the statutory technical advice of JEN in matters related to nuclear safety and radiological protection. (Control responsibilities).

Such activities can sometimes conflict with such responsibilities where nuclear safety and radiological protection considerations make it advisable not to approve a given site or project, or temporally suspend the operation of a nuclear power plant, since such a decision can obstruct the implementation of the approved National Energy Plan (weighty factors are the energy projections for the coming years, the availability of alternate sources, the balance of payments deterioration as a result of the dependency on classical energy sources and their price escalation).

JEN is therefore charged by the above-mentioned existing legislation, as an autonomous state agency directly under the Ministry of Industry and Energy, with a technical advisory function from the nuclear safety and radiological protection viewpoint, not only in the licensing process but in the control and monitoring of such matters during the construction, commissioning and operation of nuclear facilities. JEN has only executive power concerning licensing of operating personnel for such facilities.

The above notwithstanding, JEN has devoting most of its efforts and possibilities to promote the peaceful applications of nuclear energy in accordance to the mission it was originally given when it was created in 1951 (survey and mining of uranium resources, radioactive isotopes, technological research and development, personnel training). About 2% of the JEN Budget for F.Y. 1978 was invested in regulatory activities in the licensing process of NPP and radioactive facilities.

It must be finally noted that the Spanish Parliament, at the time the National Energy Plan was debated and approved at the end of last July, did vote a resolution urging the Government to implement the kind of nuclear programme required by our energy needs, while at the same time establishing the most stringent safety measures. In accordance with these principles, on the second half of October, the Government has submitted to he Parliament a Bill creating a Nuclear Safety Council, independent from the Administration which, being given the required instruments will be competent to evaluate and control the design, construction and operation of nuclear and radioactive facilities, absorbing the functions JEN has carrying out in these matters until now and whereupon JEN will be converted in a solely technological research and development institution.

At the time the present paper is being written, the above mentioned Bill is pending consideration and enactment from the Congress and the Senate. Until this Nuclear Safety Council becomes a reality and while the details of the change are worked out, perhaps not earlier than this yearend, it may be timely to briefly review the past situation so as to see what can be learnt from it.

2. STRUCTURE AND STAFF OF THE NUCLEAR SAFETY DEPARTMENT AND PERSONNEL QUALIFICATIONS

The Nuclear Energy Board (JEN) has created a technical service specialised in nuclear safety and radiological protection in order to better discharge its duties as technical counsel on such matters. This service, under the official name of Nuclear Safety Department (NSD) reports directly to the Director General of JEN and is responsible for the analysis of risks and intrinsic safety of nuclear and radioactive facilities, as well as for their supervision and control.

Currently, the NSD is organised from an administrative standpoint in five technical sections:

Evaluation Operative Unit
Inspection Operative Unit
Radiological Protection Operative Unit
Technical Standards Service
Permanent Licensing Secretariat

The functions of these sections have already been reported in the paper titled "Regulatory Inspection in Spain" submitted to

the Specialists Meeting on Regulatory Inspection Practices in Nuclear Power Plants (CSNI Report nº 27) held in Madrid in 1977 and, in substance, they are as follows:

a) Risks and intrinsic safety assessment of nuclear and radioactive facilities and issuing of Safety Evaluation Reports to be used by the Directorate General of Energy (D.G.E.) when processing applications for the different types of permits.

b) Safety inspection and risk control of nuclear and radioactive facilities during construction and operation.

c) Recommend technical standards and regulations on nuclear safety and radiological protection.

d) Participate in the process of issuing licenses to facility operative personnel.

The Nuclear Safety Department history began in 1958 when the first works on nuclear safety started in connection with the commissioning of the first Spanish research nuclear reactor - Swimming - pool-type reactor. During 1967, when the Spanish nuclear power programme was taking shape, the NSD starts a period of slowly development. To day, the Department's permanent technical staff is formed by 46 senior degree holders (engineers, scientists, etc) and 16 auxiliaries. There are also 8 senior degree holders under temporary contracts for specific work and several post-graduates under scholarships. The NSD technical staff represents only the 10% of the total JEN technical staff.

Fig. 1 shows the net permanent personnel growth and Fig. 2 the aggregate curve. The 1973-74 two year period has a perticular significance in the Department's history, since at that time the Director General of JEN placed stronger emphasis in providing the required technical staff for the Department, although a slight recession is noticed from 1978. Fig. 3 shows how the existing staff falls far below the projections made in 1971, 1974 and 1977, considering the strength and importance gradually attained by the Spanish nuclear power programme.

Table I presents the allocation of employees to technical - and administrative duties according to the responsibilities assigned to the NSD: Nuclear Facilities (power plants and fuel cycle installations); Radioactive Facilities (use of radioactive isotopes and ionizing radiation); and Site Studies.

Presently, the personnel involved in the regulatory review in the licensing process of the Spanish nuclear power plants (Design and site selection evaluation) represents in all 45% of the total NSD's manpower, and this only allows to cover less than 50% of the tasks needed by the current situation of the Spanish nuclear programme for 1979. The remaining personnel is dedicated to inspection work, technical standards preparation and nuclear safety evaluation of radioactive facilities and radioactive and nuclear materials transportation, as well as licensing the operative personnel for such facilities.

Fig. 4 represents the most significant activities of the NSD in the recent years in connection with evaluation of nuclear power plants authorisation applications, despite the personnel shortage already mentioned. It must be remembered that this same personnel is charged with the evaluation and control of other nuclear facilities in the fuel cycle field (uranium mining, second JEN Research Centre in Soria, ENUSA fuel element plant in Juzbado, etc).

The 63% of the NSD technical staff comes from other JEN Departments, where they have previously worked in nuclear research - and development. Other 30% came to work directly with the NSD under contracts or scholarships before being admitted as permanent JEN members through competitive examinations. The main stumbling block for increasing the NSD staff is this competitive examination system.

Fig. 5 presents the number of years of professional experience in the Nuclear Energy Board (JEN) of the NSD technical staff, showing an average time of 10 years. The experience in the NSD specific activities shows an average time of 5.5 years.

The technical staff shows a varied professional mixture, as corresponds to the interdisciplinary nature of the activities associated with nuclear safety and radiological protection. At this time the break down of NSD personnel by professions is as follows: 12 physicists, 12 chemists, 4 geologists, 2 mathematitians and 2 pharmacists. The engineering field is covered by 7 industrial engineers, 2 mining engineers, 2 telecommunications engineers, 1 forestry engineer and 2 technical engineers. 22% of the total are Doctorate Degree holders.

Table 2 presents the professional training of the NSD personnel in the fields of reactor technology, nuclear safety and radiological protection, not including their attending national and international experts meetings on subjects related to nuclear safety and radiological protection. Nevertheless most of the training is on the job.

3. NUCLEAR SAFETY AND RADIOLOGICAL PROTECTION REGULATORY REVIEW IN THE LICENSING PROCESS OF NPP AND EFFORTS REQUIRED.

The review of the nuclear safety of a Spanish nuclear power plant starts, from the standpoint of the Public Administration, with the processing of the several permits required and continuous throughout its entire active life. Table III shows the various stages into which this review is divided according to the regulations in force. Fig. 6 represents the typical sequence of such stages. These stages do not reflect the specific actions to be taken in extreme situation when a nuclear accident does ocurr or in the case of a mandatory facility close down.

The NSD evaluates each permit application submitted in view of the particular features of the project and of the site selected by the utility and decides whether a reasonable assurance exists

that its construction and operation will not represent an unduly hazard for human life and health and/or property. Upon finishing the review the Department submits a report to he Director General of JEN covering the safety analysis carried out and the technical conclusions arrived at. On the basis of this report, it also submits a decision proposal to the Directorate General of Energy with the recommendations required in its judgement.

The Nuclear Safety Department's experience is based on its participation in the nuclear safety assessment and inspection of:

- 3 nuclear power plants under temporary operation permit (a).
- 7 1000 Mwe Units in advanced construction process (b).
- 3 1000 Mwe Units with construction permit recently granted (c).
- 5 approx. 1000 Mwe units with preliminary permit granted and construction permit application in process (d).
- 12 preliminary permit applications in process for 19 new units (e).

Next, the most significant objectives are presented regarding the safety assessment in each regulatory stage.

3.1. Most significant technical aspects in each of the licensing process stages.

(a) - José Cabrera (Zorita): with 1 loop PWR 160 Mwe reactor, designed by Westinghouse, in service since 1968.

- Santa María de Garoña: with a BWR/3 460 MWE reactor Mark I containment, General Electric designed, in service since 1970.

- Vandellós I: with a graphite/gas, natural uranium, 480 Mwe reactor, in service since 1972. French EDF design.

(b) - 6 PWR 3 loops with NSSS designed by Westinghouse (Almaraz, Lemóniz, Ascó, 2 units each).

- 1 BWR/6 Mark III containment, designed by General Electric (Cofrentes).

(c) - 2 BWR/6, Mark III containment (Valdecaballeros).

- 1 PWR German KWU design (Trillo).

(d) - 1 PWR Ascó type, designed by Westinghouse (Vandellós 2).

- 1 PWR (Vandellós 3).

- 1 PWR, RESAR-31 Westinghouse designed (Sayago).

- 2 PWR, German KWU design (Regodola and Trillo II).

(e) - Aragón (2), Asperillo (2), Bajo Cinca (1), Cabo Cope (1), El Páramo (1), Escatrón (2), L'Ámetlla de Mar (2), Oguella (2), Punta Endata (2), Santillán (1), Tarifa (2) and Vergara (1).

3.1.1. Stage 1 (Preliminary Permit Application)

Generally, the evaluation to be carried out at this stage involves determining whether the proposed facility is or not compatible with the chosen site. Greater emphasis is placed on the study of the site suitability from the demographic, geological, seismological, hydrological and meteorological standpoints, determining on the one hand the site parameters compromising the facility which must be later taken into account during the facility design, and on the other hand, the potential effects of the facility on the environment.

The risk analysis is made using conservative assumptions where data is only preliminary or not well defined. Risk to the population must be comparable to the risks of other human activities and compared to the hazards from natural environmental radiation.

As a rule, applicants have not yet decided at this stage the type of facility to be built, so the project is not a clearly defined one. In actual practice, some basic design characteristics are assumed similar to those of other facilities either already in operation or under construction, both in Spain and in foreign countries, and this presents no particular problem if the plant to be build - is of an already known and proven type. But often the site has not been sufficiently studied and this necessitates additional research.

Evaluation lends special attention to the study of soil structures and faults and to determining the ground horizontal acceleration for the seismic design (for these cases 10CFR100 criteria have been adopted). Non-natural effects, such as an airplane crash are considered with a probabilistic approach. The proposed site is surveyed by experts in geology, meteorology and radiological protection.

3.1.2. Stage 2 (Construction Permit Application)

In the construction stage, which can be divided into a number of sub-stages, the objectives are:

1. Evaluation of the documents submitted by the applicant to support the construction application, checking that the facility has been designed in compliance with the existing nuclear legislation and with the nuclear safety and radiological protection criteria spelled out in the preliminary permit. The evaluation is centered around reafirming that the reactor specified is compatible with the site chosen.

2. Ensure that the construction is executed in accordance with the construction permit clauses.

3. Evaluate pre-operational tests and supervise their performance.

3.1.1. Stage 3 (Temporary Operation Permit)

The objectives of this stage are:

1. Evaluate the documents submitted by the applicant in support of the operation permit and the results of the preoperational tests.

2. Evaluate nuclear tests and supervise their performance.

3.1.4. Stage 4 (Definitive Operation Permit)

The temporary operation period is a preparatory period both from the technical and the administrative viewpoints, prior to the normal operation of the facility.

The definitive operation permit is to be applied for when the plant is operated as originally planned. The results of the nuclear tests programme and the development of the temporary operation are evaluated at this stage.

3.1.5. Stage 5 (Commercial Operation)

Compliance with the operation permit terms and operational incidents are evaluated at this stage. The annual fuel reloads and their affects on facility safety are also evaluated.

3.2. Required Efforts

Table IV presents the typical average efforts, expressed in man/weeks, required to complete the regulatory review of a typical light water nuclear power plant of proven type, with an electrical output of one thousand Megawatts, during the licensing process and commercial operation. These data is based on the current experience and practice of the Nuclear Safety Department. It can be deduced - from this that the regulatory review of a nuclear power plant means a total effort up to the final operation stage of 1725 man/weeks of senior degree personnel and 83 man/weeks/year thereafter during commercial operation.

4. REGULATORY REVIEW PROCEDURE

Flow chart in Fig. 7 is a schematic representation of the regulatory review procedure and formalities, from time of application (stages 1 and 2) up to the time the pertinent ministerial resolution is published in the Official State Gazette (stage 14). JEN starts acting as soon as the Provincial Delegation of the Ministry of Industry and Energy's file and report is received.

The Evaluation Operational Unit of the NSD has, among other functions, the responsibility of supervising, coordinating and carrying out the application evaluation. As a result, it is also responsible for the preparation of the technical evaluation report, containing the antecedents, results and conclusions of every analysis and studies performed in connection with the application (stage 9). Finally, the Unit submits a proposal for the technical decision on safety, based on the conclusions of such prior evaluation, together with the nuclear safety related limitations and conditions as found advisable (stage 10).

For evaluating and analysisng the technical information re-

ceived by the Evaluation Operational Unit, experts, either permanent or under time contract, are available to the Unit. Nevertheless, the matters involved are so complex and diversified that it is advisable whenever possible, to obtain experts assistance from other Units and Services of the NSD or from other JEN's Departments, who can contribute their knowledge and experience and, when required, even specialists from other outside competent National Institutions may be summoned to collaborate, such as the National Meteorological Institute and the Department of Seismology and Seismic Engineering of the National Geographic Institute.

The nuclear safety assessment of a given project requires knowledge and application of:

a) Criteria, codes, guidelines and standards existing on the subject.
b) Calculation models and particular design programmes for predicting and analysing the effects of the facility operation under normal conditions and transients and foreseen nuclear accidents.
c) Basic technology of the reactor type used in design.
d) Particular parameters for the chosen site and for the potentially affected environment.
e) The experience gained from similar type nuclear plants, either in actual operation or under construction.
f) Results of research and development programmes on nuclear safety for the reactor type used in the design.

The quick pace of development of nuclear technology and of nuclear safety and radiological protection philosophies required a considerable follow-up and up-dating effort.

It is then essential to maintain close collaboration and information exchange activities with international agencies involved in this field, such as OIEA and OECD, and with other countries' authorities and regulatory agencies, particularly those where the technology for the plant to be built in Spain originates from.

When necessary, clarification of documents submitted or additional information deemed to be important with regard to nuclear safety and radiological protection (stage 6) are requested from the applicant.

It is the responsibility of the applicant to submit as many clear and accurate data and studies as are required by virtue of the RINR provisions, and he is entitled to present directly to JEN the revisions or changes to the original documents as they are necessary and all these addenda become a part of the relevant file. Any clarification, study or additional report requested will be to the charge of the applicant.

After the Safety Evaluation Report is completed and the technical decision proposal is prepared with the pertinent nuclear safety conditions (stage 10), both documents are referred to other Operational Units and Services of the Nuclear Safety Department for

their consideration and suggestions when appropriate. Later a meeting is held with the applicant to inform him of the decisions arrived at where the applicant is allowed to express his comments - (stage 11) or other technical changes may be agreed on.

Finally, the Department Director submits the safety recomendation and the prepared Safety Evaluation Report (stage 12) to the Director General of JEN. After his approval these documents are send to the Directorate General of Energy for processing (stage 13).

4.1. Project Manager

A project manager is appointed to coordinate the evaluation activities for each individual application. The project manager must have an overall awareness of the project and of the nuclear safety problems involved and must be up-to-date on the development of such problems and on the applicable safety and radiological protection - criteria, not to forget a general knowledge of the various regulations existing.

The Project manager interprets- or adopts - the technical - viewpoints and criteria contained in the reports, studies and evaluations conducted by the Department experts or consultants, reporting to his Unit Head any disagreement he may have with such findings.

4.2. Evaluators and Experts

The members of the Evaluation Operational Unit, whether they are permanent, on a time contract or temporarily transferred from other units and the specialist from other Department Units or Services to whom specific tasks may have been assignaed in connection to the evaluation of one given application, must prepare a report on the results of the study or evaluation of the subject assigned them and submit it to the Project Manager (stage 8).

All work subjects or project aspects requiring evaluation - are to be assigned to Department evaluators or experts. Each evaluator or expert studies and reviews the information concerning his assigned subject and decides upon the necessity or convenience of consultation with others after he has studied the matter on a preliminary basis. Although calling consultants is an acceptable procedure at any time of the evaluation process, it is preferable to summon their advice at the early stages of evaluation.

The consultant request form is clear and specific so the consulted advisor can quickly identify the problem brought before him and solve it effectively and in a short time.

4.3. Evaluation Plan

Before any work is started an Evaluation Plan must be drawn. This plan is to specify the distribution of the different aspects requiring an analysis, starting dates for all tasks and partial evaluation report filing dates for each participant and a final completion date. Under no circunstance any attempt to meet these dates

can result in a loss of quality of the review work which must be, in every case, as deep and extensive as necessary.

The plan must also specify the criteria and procedures to be followed in each case with references made to the applicable Department internal documents and guidelines. To this aim, the Nuclear - Safety Department is now preparing specific guidelines for the evaluation of the different permit applications, in order to standardise and facilitate the work of evaluators and Project Managers. Table V presents a list of contents of one of such guidelines.

5. EXAMPLES OF PROBLEMS ASSOCIATED TO THE REGULATORY REVIEW

5.1. Problems derived from the technical-administrative licensing process

The Spanish nuclear power plant programme history can be divided in three periods since its inception. The incorporation in 1972 of the Regulations on Nuclear and Radioactive Facilities (RINR) affected in a different way each of these periods. The clarifying function of this document regarding the licensing process of a nuclear power plant is essential when analysing past experiences.

a) The first period covers the three plants now in operation, all of then approved before RINR enactment.

b) The seven units now under construction at four separate sites make up the second period. All their preliminary permits were granted before RINR enactment, bur the respective construction permit was absolutely under the influence of the aforementioned Regulation.

c) Lastly, the third period covers all plants with their complete licensing procedure totally governed by RINR.

The administrative procedures for authorisation granting were developed during the first period, as first reflected by a regulation draft which was used as a basic guideline. Once the legal - framework was set, the following years -as stated before- were used to complement it by adopting internationally recognised guidelines, codes and standards, as well as explicit Administration statements on technical criteria through the permit conditions.

We have thus arrived to the present time, where it can be - said that a technical -administrative base exists with sufficient definition to make possible- setting aside the particular aspects of each project - the search for a time perspective of the entire authorisation process with the necessary reliability.

Table IV presents the times used in the processing and granting of the Preliminary and Construction Permits for plants included in period b) and some included in period c) for which Preliminary Permit has been only recently granted. Times used in processing the permits for plants already operative are not presented,

since these are not deemed to be significant. In considering the data in the Table, no clear conclusions can be derived, although it is possible to reason about them.

Since the scope and depth of the documentation to be submitted by the applicant in the Preliminary Permit stage is now better defined, it is foreseeable a drastic reduction of the time required by the Administration for reviewing the information, avoiding delays and inactive periods imposed when it is necessary to request additional information from the applicant. In principle, and only as a guideline, the following times may be considered: Documents - preparation by applicant: 12-14 months; preliminary permit procedure: 9-12 months.

Regarding the Construction Permit, two observations are pertinent. The first one relates to the greater accuracy now available as to the documents to be submitted by the applicant, with the effects mentioned above. The second one is determined by the fact - that, according to RINR, the granting of the preliminary permit implies the approval ot the site, meaning that all site studies have been satisfactorily completed at this early stage. All the above - allows to state that the Construction Permit process can take fifteen months, approximately.

5.2. Problems Derived from Project Diversification

The nuclear safety regulatory review in a nuclear programme such as the Spanish, becomes extremely complicated owing to the use of different technologies, caused by the utilities taking recourse to the international market in search for the best offers at each time.

Although the three nuclear plants now in operation in Spain were designed, built and authorised before the now existing legislation was applicable, they are a good example of project diversification; their nuclear systems are all different, namely: PWR, BWR and graphite-gas; their powers vary widely: 160, 440 and 460 Mw; different site conditions: two on a river bank and one by the ocean coast; and different technology origin: two American and one French.

The strict use in these early experiences (both by the Public Administration and the utilities) of the "turn key plant" concept, greatly facilitated the licensing process.

The following plant generation, all with American NSS systems (6PWR and 1 BWR) used the "reference plant" concept in a flexible way; there are appreciable differences in the overall design of the three twin PWR type plants, although all of them use the same American plant as the reference Plant. Certainly these differences do not substantially affect the nuclear system (NSSS) nor - the technological safeguards, but aspects such as general layout, radwaste systems, etc., are not necessarily alike among these three plants, nor with respect to the reference plant (BOP).

In this case the licensing process requires a considerably greater effort, both by the Administration and by the utility: detailed systems and equipment descriptions, questions and answers, explanation of differents technical solutions from those used in the Reference Plant, etc. But these have lead both parties involved to build up an experience that probably could not have been possible with the previous procedure.

The passing from one version of the reference plant concept to another version has been accompanied by a notable increase in - each unit power and, at the same time, by an enormous proliferation of safety criteria, standards and requirements to follow; suffice to mention in this regarg features related to accident prevention, seismic structural design, reduction of permissible radioactive discharges, etc. It is not necessary to underline the help provided by the easy access to information and criteria afforded by the supplying countries.

Until now, except Vandellós (gas-graphite) all nuclear power plants in operation or under construction have American originated nuclear systems. American criteria and standards are applied to these plants in the manner or up to the time stipulated by the Administration after consultation with the utility. This has facilitated -within the complexity of each case and despite the variations from one case to the next-the establishing of common technological groundwork.

The supply of nuclear systems from countries other than USA (although similar to the PWR's and BWR's now under construction in Spain), adds a new complexity facet to the tasks of the Administration and of utilities since there are not only differences in the systems, but also in the criteria used to evaluate the entire plant. A recent experience of this kind has been the evaluation of the Trillo power plant (PWR by KWU) application.

Site choice, power, reactor type, country of origin, dates of applicable criteria, design engineer, etc., all are variables in each plant and for this reason authorisations can only be processed on a case by case basis.

A solution being prepared for the near future is to standardise projects, a concept including the previous planning of nationwide available sites.

5.3. Problems Derived from NSD's Manpower Shortage

The scarcity of manpower available to the NSD is one of the largest stumbling blocks for the implementation of the Spanish nuclear programme within the set times due to the fact that the accumulation of evaluation and control tasks makes it difficult for the Department to perform its responsibilities in the field of nuclear safety and radiological protection. The commissioning of the 7 units presently under advanced construction is a top priority which will make it unfeasible for the NSD to start evaluating new projects.

All the above has been newly complicated with the recent Three Mile Island incident which has forced the Department to initiate a study on its consequences and possible implications in Spanish nuclear power plants.

Using experts from other JEN Departments has not solved the problem, as shown by past experience. The main difficulties encountered in using such experts are:

a) Lack of specific knowledge on the state of the art in the nuclear safety and radiological protection fields and particularly insufficient awareness of the typical problems in this area, of the evolution of applicable criteria and lack of up-dating on the abundant technical reference information. This requires time to retrain and recondition such experts. People with research background are inable to perform regulatory review duties in most cases.

b) The large volume of work involved requires in most cases this work to be done on a full time basis, which is a virtually incompatible proposition in view of the other duties that must be discharged in the normal job of the experts involved.

c) The administrative stiffness tend to lower the quality of the work produced since the experts are prevented from becoming integrated into the evaluation task force.

d) Lack of time to write specifications and conditions for the required consultations and to exercise a degree of control, given the disproportion existing between the amount of work to be done and the available qualified personnel. The result can be that brilliantly performed work is rendered useless in actual practice from the licensing standpoint.

e) The lack of qualified experts in certain areas within JEN itself. This has been solved using outside experts, both domestic (Engineering firms offering sufficient assurence of independent judgement) and foreign. The above limitations are forcing a growing use of outside consultants.

The present administrative status of JEN, subject to the rigid regulations for State Agencies autonomous from the Central Public Administration, prevents new qualified individual from joining JEN, since an increase in the permanent payroll is extremely dependent on budgetary considerations. All personnel must be admitted through a competitive examination system leading to their becoming permanently engaged as Civil Servant and this recruiting procedure causes enormous difficulties and waste of time. Direct time contracting is no an easy solution either, since the contracts are limited by law to a short time and cannot be extended.

6. CONCLUSIONS

The following conclusions can be extracted from all the foregoing:

6.1. Legal Reform

The existing Spanish legislation in basically administrative in nature and contains practically no technical criteria or standards. But, although nobody will deny its usefulness and the contribution it has made in achieving the present state of Spanish nuclear development, it has fallen far behind from the legislation - enacted elsewhere during the recent years. The experience gathered suggests it is time to revise the existing legislation on nuclear matters.

This reform should contemplate the full independence of the agency responsible for nuclear safety and radiological protection from the Central Public Administration departments charged with the promotion and planning of the national energy resources. It seems that the creation of a Nuclear Safety Council will help to solve - this problem.

6.2. Reinforcement of the Agency Responsible for Nuclear Safety

Nuclear Safety regulatory review and inspection must be reinforced with sufficient personnel in view of the size of the Spanish nuclear programme and of the public opinion preocupation with the safety afforded by the programme. Research and specific development programmes on nuclear safety aspects must be implemented, besides arranging for co-operation with other similar programmes currently in progress in other countries, particularly concerning light water reactors. Nuclear safety regulatory activities and research activities must be concordant and co-ordinated in order to better - perform their responsabilities and to mutually convey their needs and knowledge. Promotion activities must not interfere with their independent criteria and responsabilities.

6.3. Public Information

The existing procedures to convey information to third parties and to the public at large must be improved, adopting adequate methods and channels. Free access to files and required data must be provided, although keeping confidential all matters affecting national security or industrial rights. Issuing periodic reports not only to the media but to the Government and political circles can be useful. These releases should contain the nuclear plants operation results, progress of the nuclear programme, troubles or accidents with a safety significance. This information must be objective, immediate and , whenever possible, disclosed before the matter is brought to public attention by other means. This will help to - keep the people informed on what is being done and in gaining the necessary public confidence.

6.4. Site Policy

One of the permits requiring longer time to be granted is - the preliminary site selection approval. It would be a good idea to carry out a deep study of the national geography, whether at the

utility company level, utilities association, regional bodies or central government levels, to define the areas more suitable for nuclear plant installation, even pinpointing specific sites for potential use. Thus, utilities will be able to request with sufficient advance time the sites they need to develop in the next 5 or 10 years. The most outstanding consequences of this policy will be to reserve areas of specific sites within the Spanish territory so these can be taken into account during future territorial planning. In any event, the need is felt to change the current site selection policy, giving a strongest part to local authorities and regional bodies in territorial development, area selection or in setting - aside specific sites where nuclear plants may be built, and to this effect, each application will take into consideration alternative sites.

The above conclusions will probably be taken care of and solved in a greater or lesser degree in the near future with the implementation of the Resolution on Nuclear Energy approved by the Congress during the meeting held on 27th/28th July of this year.

TABLE I

FUNCTIONAL ASSIGNMENT OF NSD TECHNICAL PERSONNEL

	MANAGEMENT	ASSESMENT UNIT	INSPECTION UNIT	RADIOLOGICAL PROTECTION UNIT	REGULATORY GUIDES	OPERATING PERSONNEL LICENSES	TOTAL
	DIRECTOR OF NUCLEAR SAFETY DEPARTMENT						1
		1	1	1	1		4
NUCLEAR INSTALLATIONS		9 / 2	7 / 3	5 / 1	3 / 1		24 / 7
RADIOACTIVE INSTALLATIONS		6	4	1	1	1	12 / 1
SITING		5					5
TOTAL		21 / 2	12 / 3	6 / 2	5 / 1	1	46 / 8

STAFF MEMBERS → [/] ← WITH TEMPORARY CONTRACT

SECRETARIAL AND AUXILIARY PERSONNEL 16

OCTOBER 1979

T A B L E II

NUCLEAR BACKGROUND AND TRAINING OF NSD PERSONNEL

1. NUCLEAR ENGINEERING COURSES IN SPAIN.	16
2. TRAINING COURSES ON SPECIFIC TOPICS RELATED TO NUCLEAR SAFETY IN SPAIN.	11
3. NUCLEAR SAFETY AND RADIOLOGICAL PROTECTION COURSES AT HARWELL (U.K).	13
4. I.A.E.A. NUCLEAR SAFETY COURSES.	1
5. ONE YEAR OF WORKING EXPERIENCE WITH THE NRC STAFF (U.S.A).	3
6. NRC COURSES ON LICENSING, NUCLEAR SAFETY AND REACTOR TECHNOLOGY.	7
7. WORKING EXPERIENCE ABROAD WITH VENDORS, UTILITIES, CONSULTANTS AND REGULATORY BODIES.	16
8. NUCLEAR SAFETY COURSES ABROAD AT UNIVERSITIES OR ENGINEERING COLLEGES.	3

- - - - - - - - - -

TABLE III

REGULATORY REVIEW STEPS IN THE LICENSING PROCESS OF A NUCLEAR POWER PLANT AND RELATED ASSESMENT FUNCTIONS

STEP 1.- Review of the documents presented by the applicant for the Preliminary Permit (site description and objetive). Safety Evaluation Report and proposal for Limits and Conditions.

STEP 2.- 2a)
Review of the Proyect and Preliminary Safety Analysis Report presented in support of the application for the Construction Permit. Safety Evaluation Report and proposal for Limits and Conditions.

2b)
Safety review during construction.

2c)
Preoperational tests.

STEP 3.- 3a)
Review of the documents (1) presented in support of the application for the Temporary Operation Permit. Safety Evaluation Report and proposal for Limits and Conditions.

3b)
Initial Startup tests.

STEP 4.- Review of the application for the Definitive Operation Permit.

STEP 5.- Review of the plant operation and fuel reloads.

---

(1) Final Safety Analysis Report, Technical Specifications, Initial Startup tests, Results of the preoperational test, Emergency Plans and Operational Regulations.

TABLE IV

TYPICAL EFFORT (WEEKS x MAN) NEEDED TO COMPLETE THE NUCLEAR SAFETY AND RADIOLOGICAL PROTECTION ASSESMENT OF A NPP ACCORDING THE SPANISH LICENSING PROCESS (CASE OF PROVEN TYPE).

	PRELIMINARY PERMIT	CONSTRUCTION PERMIT / CONSTRUCTION	TEMPORARY OPERATION PERMIT	DEFINITIVE OPERATION PERMIT	NORMAL OPERATION (PER YEAR)
METEOROLOGY, HYDROLOGY, SEISMOLOGY, GEOLOGY AND GEOTECHNIC	170	80	20		10
DEMOGRAFY, ECOLOGY AND NEARBY SITE ACTIVITIES	60	40	20		
DESIGN CRITERIA AND REGULATORY GUIDES	6	100	10		
REACTOR PHYSICS, THERMOHYDRAULICS AND FUEL		30	55		
MATERIALS		25	15		
INSTRUMENTATION AND CONTROL		30	35		
ANTISEISMIC DESIGN AND STRUCTURAL ANALYSIS		40	15		
CONTAINMENT		20	25		
ELECTRICAL SYSTEMS		15	12		
AUXILIARY SYSTEMS		20	25		
ENGINEERING SAFETY FEATURES		23	25		
SHIELDING		20	16		
RAD WASTE SYSTEMS		15	24		
RADIOLOGICAL PROTECTION, MONITORING AND SURVEILLANCE PROGRAMS	20	25	8		
ACCIDENT ANALYSES		40	50		30
QUALITY ASSURANCE AND APPLICANT ORGANIZATION	3	20	8		
OPERATIONAL REGULATIONS			5		
TECHNICAL SPECIFICATIONS			20		
EMERGENCY PLANS			5		
PREOPERATIONAL TESTS		24			
INITIAL STARTUP TESTS			55	8	
PERIODICAL REPORTS AND REVIEW DURING CONSTRUCTION		98			
DESIGN CHANGES AND ADDITIONAL INFORMATION		250		10	
NORMAL OPERATION CONTROL				8	
RELOADS					43
QUESTIONS AND SAFETY EVALUATION REPORTS	14	25	34	4	
TOTAL	273	940	482	30	83

T A B L E V

TYPICAL INDEX OF AN EVALUATION PLAN FOR A PARTICULAR PERMIT APPLICATION

1. INTRODUCTION.
2. NUCLEAR SAFETY DEPARTMENT RESPONSIBILITIES FOR THE ASSESMENT OF DE PERMIT APPLICATION.
3. ORGANIZATION.
4. PROYECT LEADER MISIONS.
5. STAFF AND CONSULTANTS PARTICIPATION.
6. EXTERNAL CONSULTANTS AND EXPERTICE FROM OTHER BODIES.
7. ASSESMENT PLANNING.
8. TECHNICAL CRITERIA.
9. PROCEDURES AND METHODOLOGY.
10. ADMINISTRATIVE ASPECTS.

- - - - - - - - - - -

T A B L E VI

TIME BETWEEN AN APPLICATION FOR A PERMIT AND THE ISSUING OF THE PERMIT BY THE DIRECTOR GENERAL FOR ENERGY

PRELIMINARY PERMIT

	DATE		
	APPLICATION	ISSUING	MONTHS
ALMARAZ	JUN/69	OCT/71	28
ALMARAZ (1).....	JUL/71	MAY/72	10
LEMONIZ	DEC/67	FEB/69	14
LEMONIZ (1).....	DEC/71	MAY/72	5
ASCO	JUL/70	APR/72	21
COFRENTES	DEC/71	NOV/72	11
TRILLO (2).....	MAY/72	SEP/75	40
SAYAGO	SEP/73	SEP/75	23
VALDECABALLEROS	MAY/74	SEP/75	16
REGODOLA	NOV/73	AGU/76	21
VANDELLOS II Y III ...	MAY/74	FEB/76	21
ESCATRON	MAR/74	APR/77 (3)	

CONSTRUCTION PERMIT

	DATE		
	APPLICATION	ISSUING	MONTHS
ALMARAZ I	JUL/72	JUL/73	12
ALMARAZ II	JUL/72	JUL/73	12
LEMONIZ I	SEP/72	MAR/74	18
LEMONIZ II	SEP/72	MAR/74	18
ASCO I	DEC/72	MAY/74	17
ASCO II	DEC/73	MAR/75	15
COFRENTES	FEB/74	SEP/75	19
VALDECABALLEROS I Y II	DEC/75	APR/77 (3)	
TRILLO	SEP/76	MAR/78 (3)	

NOTES

(1) New application for higher electrical power.

(2) Site and electrical power changes during licensing process.

(3) Safety Evaluation Report by JEN.

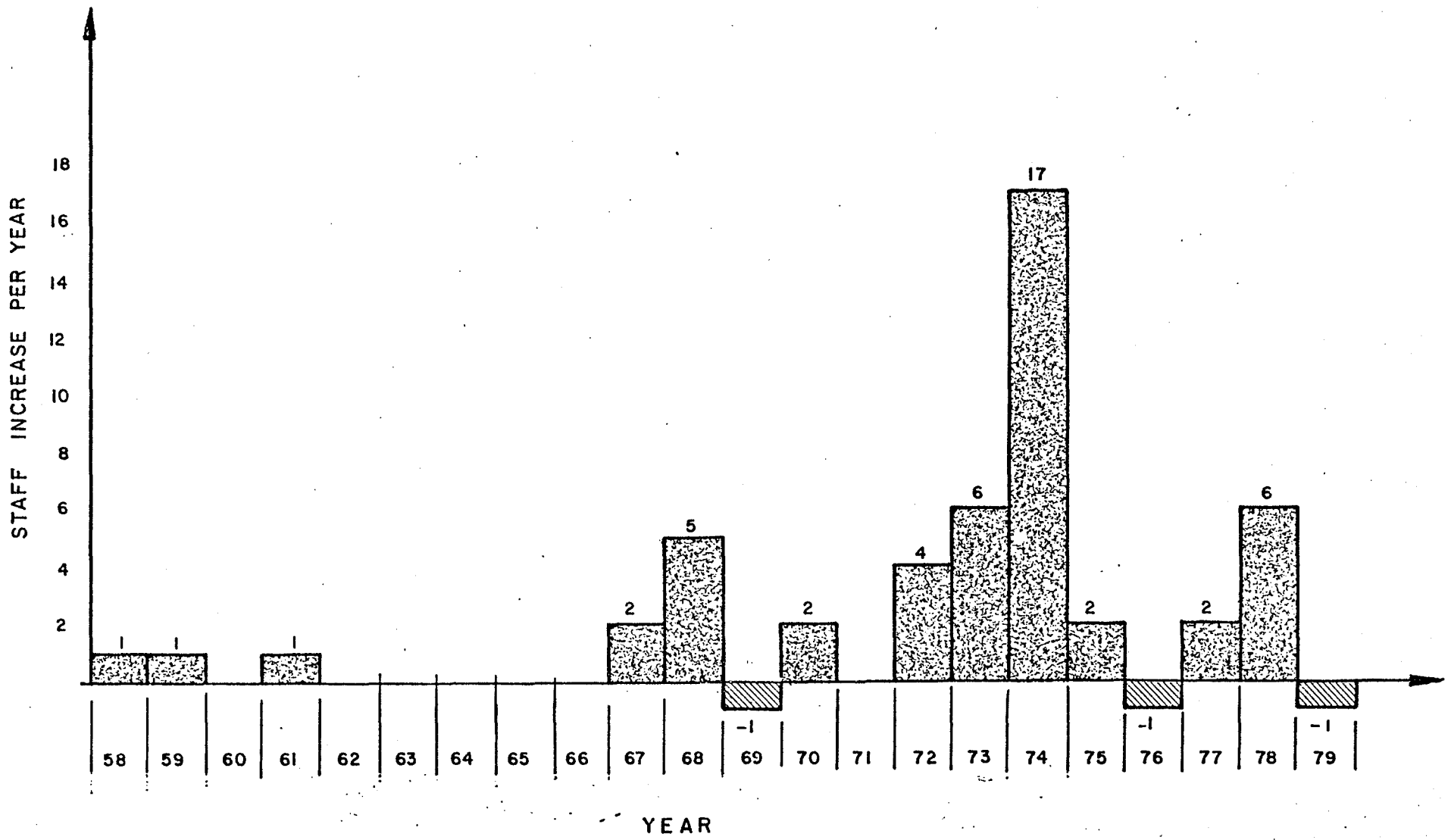


FIG.1.— NUCLEAR SAFETY DEPARTMENT PROFESSIONAL STAFF NET INCREASE PER YEAR.

DSN/3006a/1079

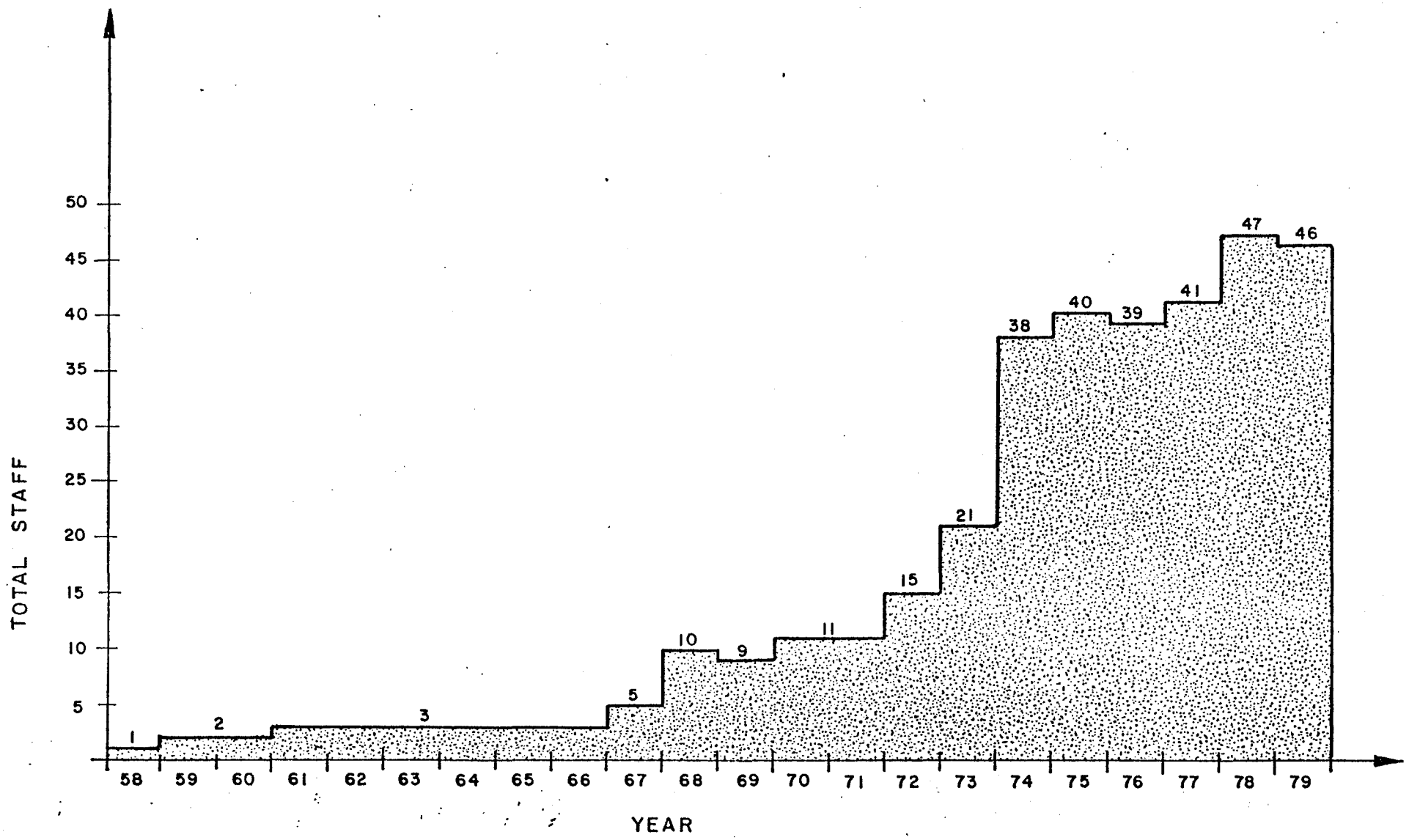


FIG. 2.— NUCLEAR SAFETY DEPARTMENT PROFESSIONAL STAFF EVOLUTION.

DSN/3007a/1079

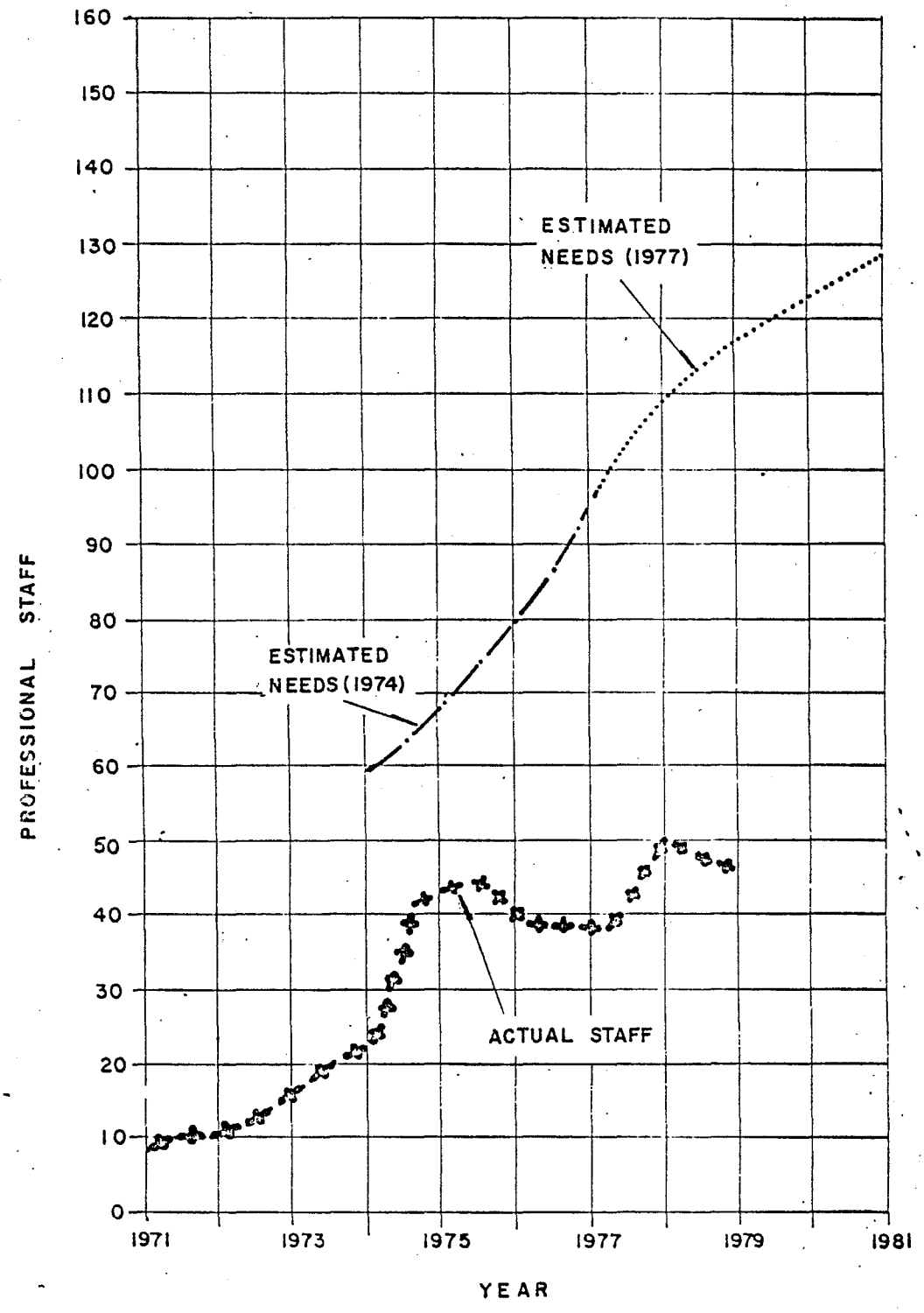


FIG.3.- NUCLEAR SAFETY DEPARTMENT PROFESSIONAL STAFF EVOLUTION.

DSN/1115a/1079

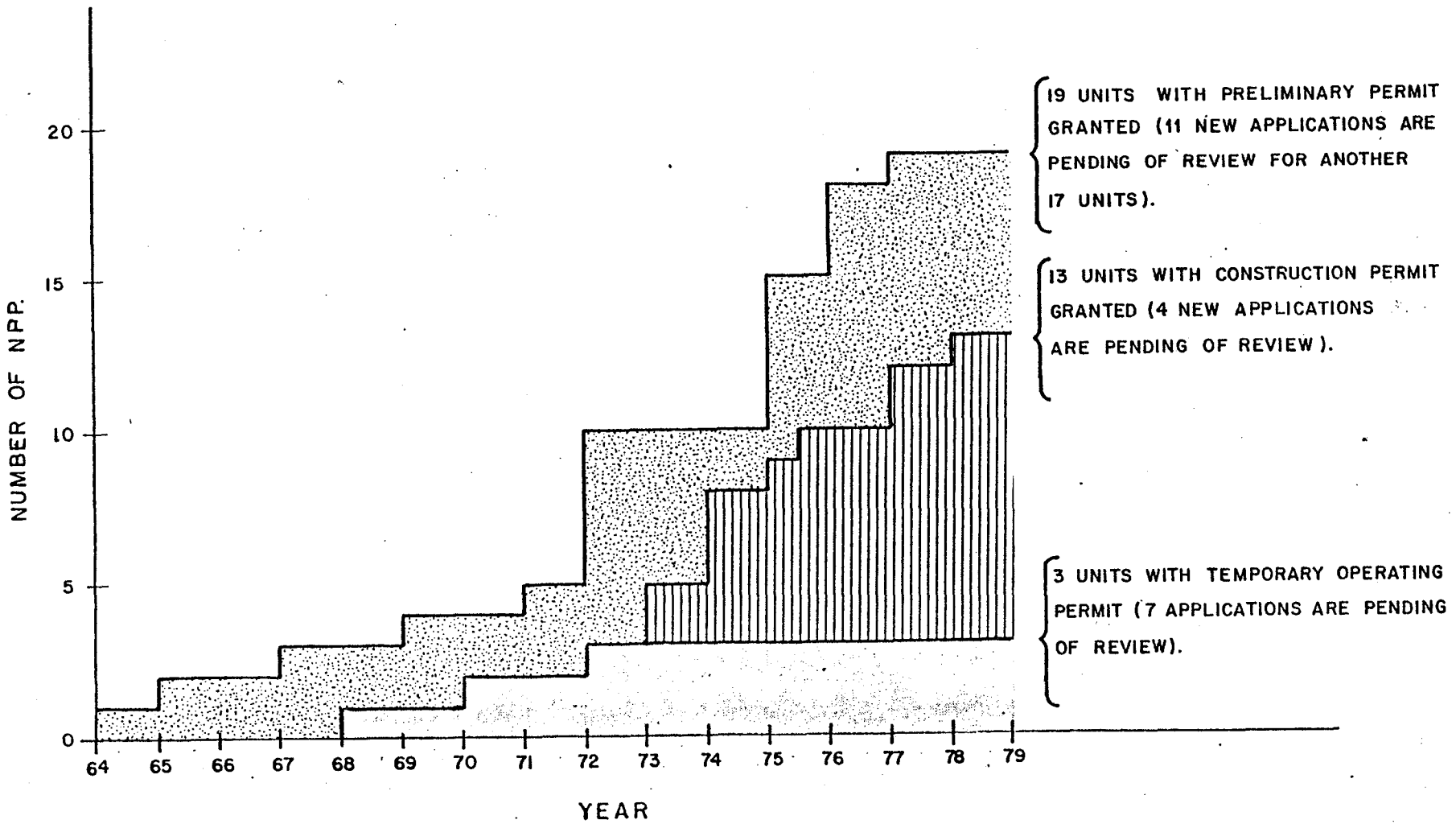


FIG.4.— NPP REVIEWED BY THE NUCLEAR SAFETY DEPARTMENT.

DSN/3008a/1079

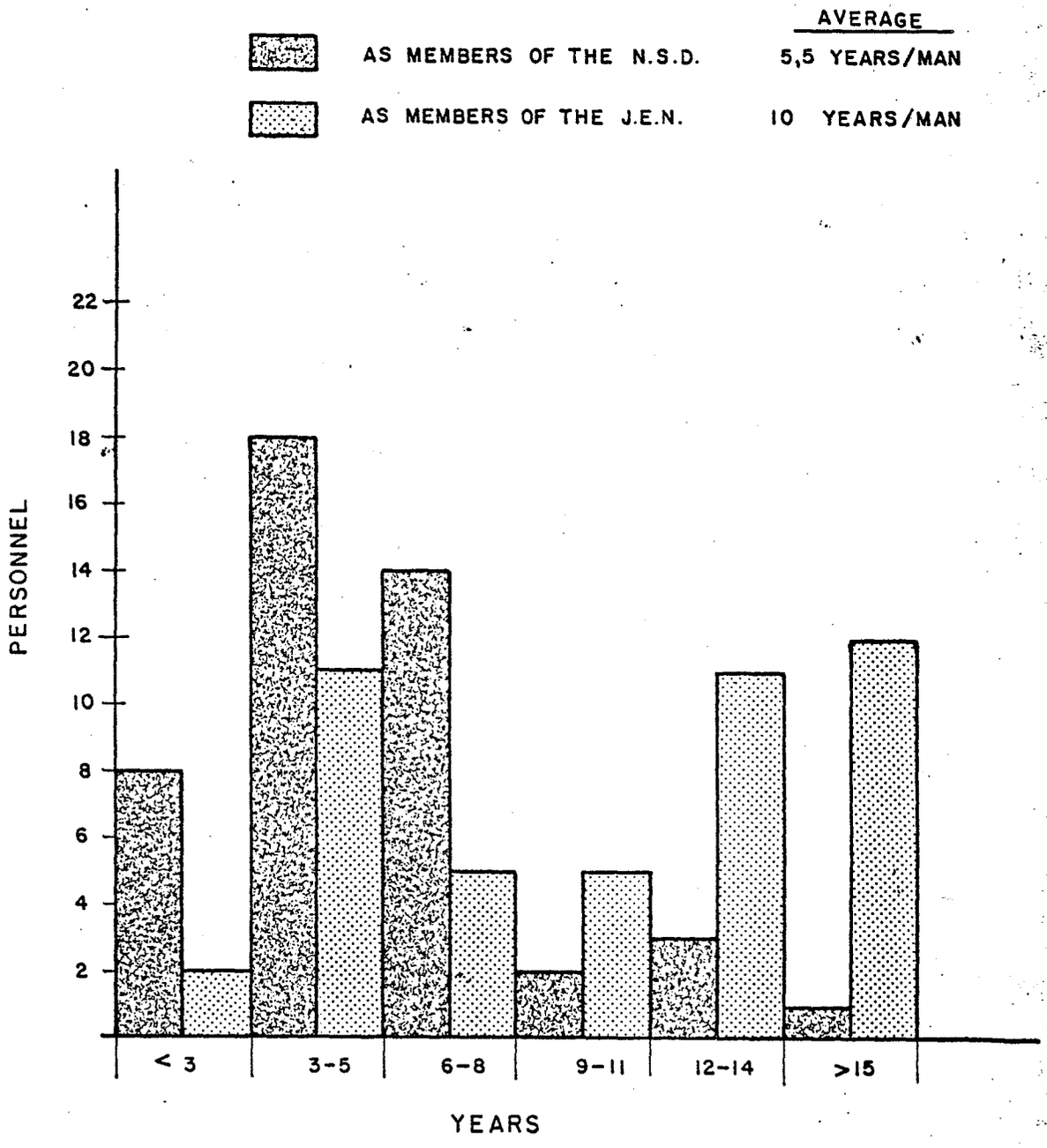


FIG.5.– WORKING EXPERIENCE IN YEARS OF THE NUCLEAR SAFETY DEPARTMENT STAFF.

DSN/3009a/1079

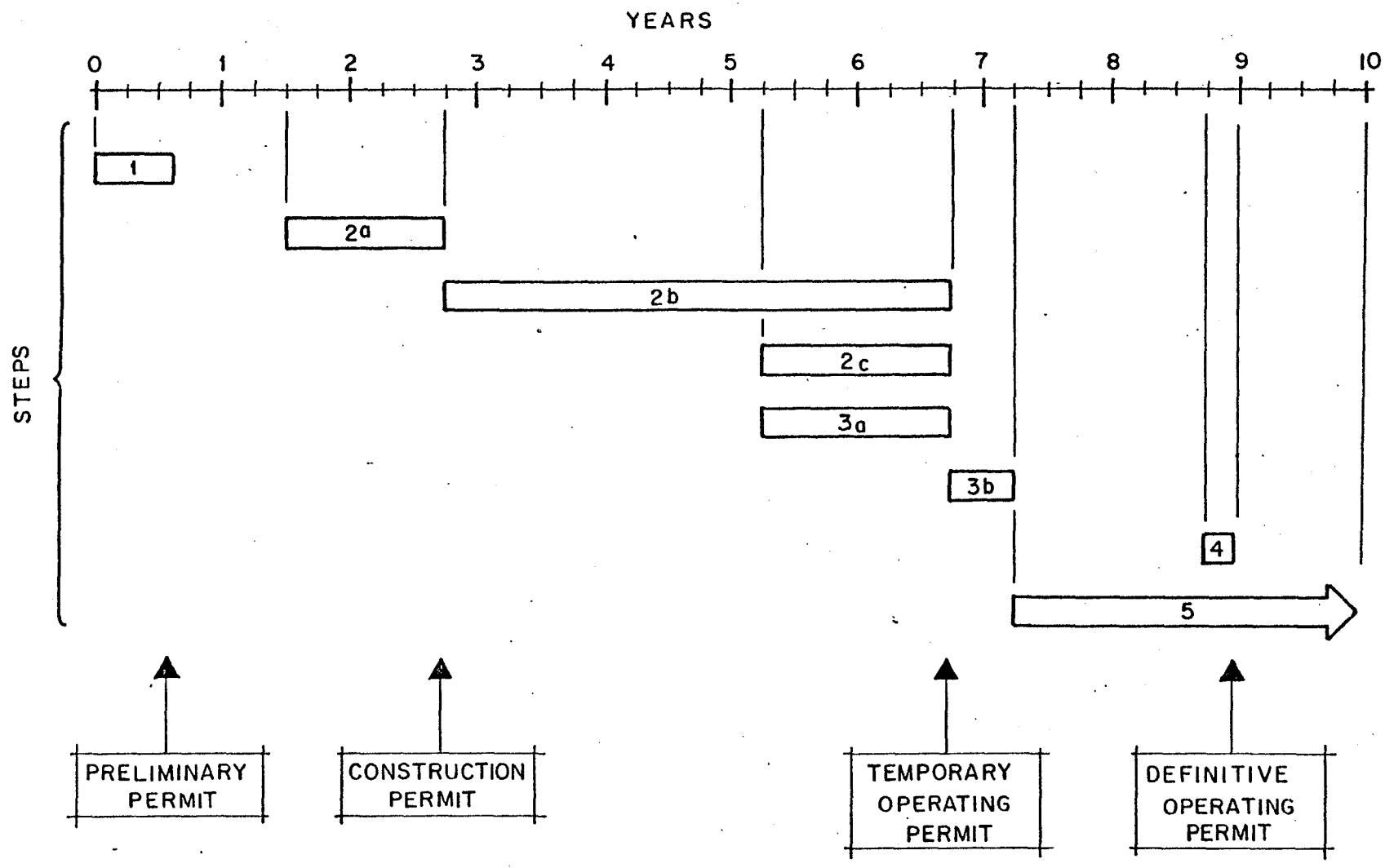


FIG.6.—SIGNIFICANT STEPS DURING THE LICENSING PROCESS.

DSN/3010a/1079

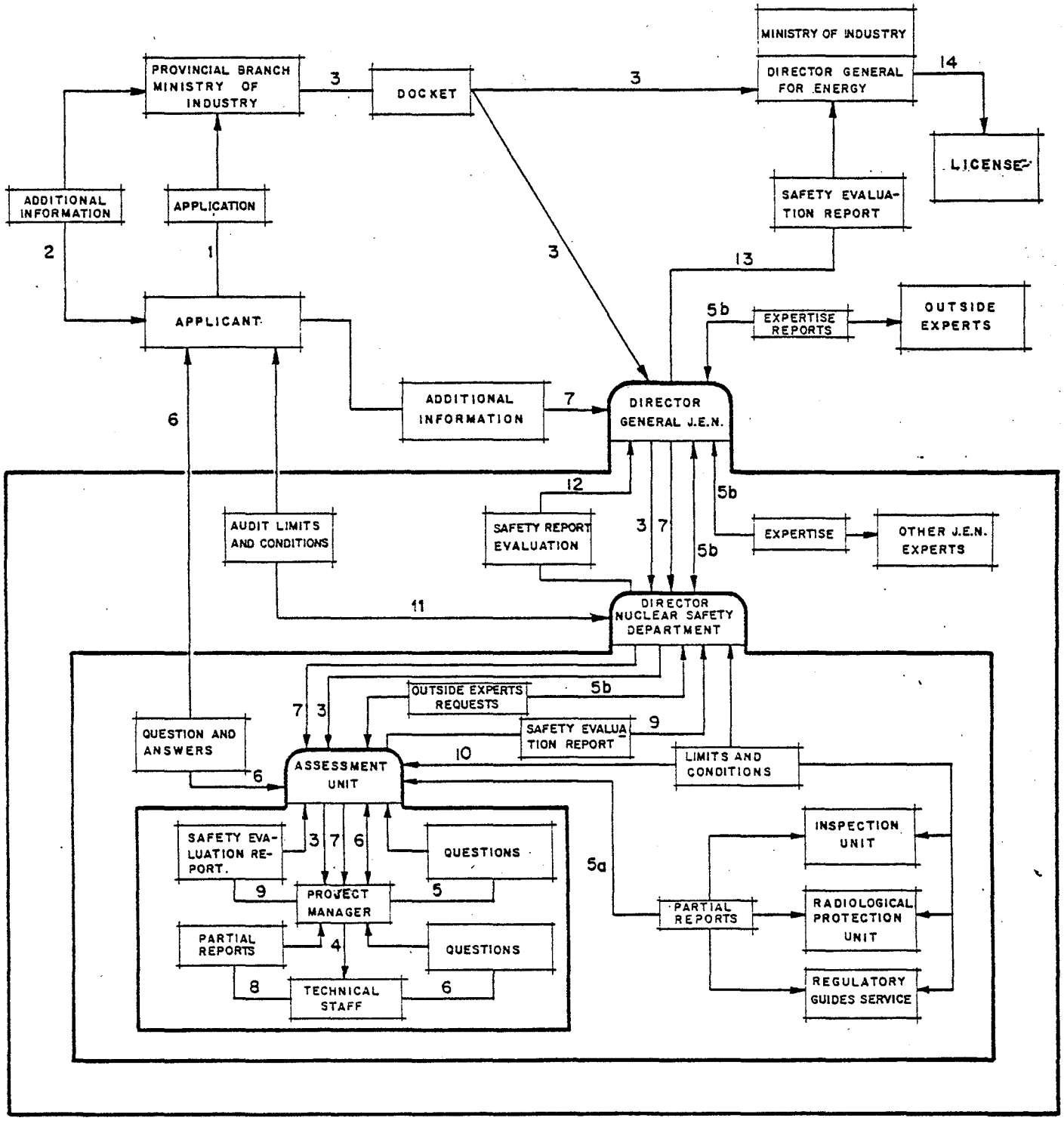


FIG.7.—REGULATORY, REVIEW PROCEDURE IN THE LICENSING PROCESS FOR NPP.

DSN/3005a/1079

First Period of Discussions on Session II

J. Edwards (UK) (To R. Gausden)
Would Mr. Gausden explain the relationship existing between the NII, the customer and site licence (e.g. the CEGB), and the industrial organization which undertakes the design and building of the nuclear station. At what stage is the executive authority of the NII exerted to influence design etc. - is it at the Stage A Safety Report stage and/or Stage B Safety Report, or can the NII direct changes at any stage? Presumably responsibility for the design and its acceptance must rest with the vendor and the ultimate owner?

R. Gausden (UK)
It is clear that until a site license has been issued there is no statutory power for the NII to take action to change design or impose requirements. However both the industrial organisation and the prospective licensee recognise that they need to clear their design concept and preliminary fault analysis before they can expect a license to be issued, so in effect the NII can, and do, influence the design from the initial conceptual stages.

J. Edwards (UK) (To C. Pérez del Moral)
All of Spain's nuclear power systems are basically designed by vendors from other countries e.g. Westinghouse and G.E. in the USA etc. To what extent does the Spanish Safety Authorities analysis of these systems have to accept the vendors' own safety analysis? Have the Spanish authorities sufficient manpower to conduct their own total analysis of such systems, are they in a position to question the vendors' analyses and safety justifications, and are they satisfied with the response they get, and the method of resolving any differences that arise? It would be helpful if the main problems that have arisen could be defined, and the solutions reached explained.

C. Pérez del Moral (Spain)
At the moment we haven't enough calculation models or technological knowledge to carry out an independent analysis in all areas relating to nuclear safety. We have developed calculation models of our own only on the subject of radiological impact on the environment under both normal operating and accident conditions, and also in some minor areas such as

spent fuel pool cooling. In addition there are some calculation codes in the field of nuclear and mechanical design and functional analysis of the containment after a LOCA. On the other hand, there isn't at the moment enough manpower on hand to continue developing these methods of analysis. What we have been doing is comparing the hypotheses, models and results submitted to our Organization against those carried out and accepted by the supplier country. In any case we try to get satisfactory answers by establishing the necessary contacts with the vendors themselves, and with the regulatory authorities in the supplier country of the Spanish reactors, using to this effect the agreements between the two governments in the matter of promoting and exchanging technical information. The paper by Mr. Brincones et al. which will be presented in Session IV explains some of the problems you refer to, and perhaps your question will be more fully answered then.

F.J. Turvey (Ireland) (To C. Pérez del Moral)
Looking back on the last 15 years of nuclear regulation in Spain do you now think that it would have helped nuclear safety if the JEN had limited the number of reactor designs allowed to be introduced during the above period by the utilities?

C. Pérez del Moral (Spain)
The answer is affirmative as will be seen from the comments on the paper presented. I believe variety of projects goes against thorough technological knowledge and therefore against the safety of the nuclear programme. The problem is naturally compounded when there aren't enough means or personnel. At the moment it does not look as though this problem can be solved in Spain since the electrical companies freely choose whatever suits them most in the market.

P. Giuliani (Italy) (to C. Pérez del Moral)
At the end of your paper you mention the possibility of doing a general site survey in your country. I would like to know if you have a plan for doing this; if so, can we have some details. I am asking this because in my country we, as a Regulatory Body, are requested by law to do such a survey. We have just completed the main stage of it and it has been a long and complex work.

C. Pérez del Moral (Spain)
At the moment there is no regulation in our country which compels the Junta de Energía Nuclear to explore the nation's territory in search of the best sites for nuclear power plants. So far the sites have been chosen and proposed by the electrical companies with the Administration merely giving its approval or setting down condition. In the new legislation that will come into effect as a result of the Resolution approved by the Congress last July, it is provided that the new Nuclear Safety Council will have the task of selecting future sites, in cooperation with the local or autonomous authorities established after the recent reorganization of the Spanish territory. The experience of other countries like Italy, which has similar programmes governing the choice of sites, will be of great value for us.

A. Kraut (Germany) (to C. Pérez del Moral)
What will be the consequences in connection with the new organization of the regulatory and licensing body in Spain concerning coming licensing procedures?

C. Pérez del Moral (Spain)
Right now it's really difficult to answer this question as the bill for the creation of a Nuclear Safety Council has not yet been discussed and approved by the Spanish Parliament, so no one knows what the final version will be like. It looks like this project will not substantially modify the present procedures, but this is still unclear. As soon as we know the details of its practical application we will be able to answer more fully.

N. Aybers (Turkey) (To R. Gausden)
Is there any possibility of granting partial permission to start some works during the construction license?

R. Gausden (UK)
The license which is issued is a site license to which conditions are added during the various stages of construction, commissioning and operation of the plant. Various 'stop' points are introduced which would prevent, for instance, any stage of construction proceeding until sufficient information had been presented and an adequate safety analysis had been carried out.

PROCEDURES D'AUTORISATION ET STANDARDISATION
PAR JM. OURY (1)

==========

1. INTRODUCTION : LE PROGRAMME NUCLEAIRE FRANCAIS

La France s'est engagée en 1974 dans la réalisation d'un grand programme électronucléaire fondé sur des tranches nucléaires de 900 MWe, comportant un réacteur à eau sous pression et uranium enrichi. Les premières tranches de ce programme sont maintenant complétement construites : six tranches de 900 MWe sont actuellement en fonctionnement (deux à Fessenheim et quatre à Bugey) : l'achèvement de près d'une trentaine d'autres (très semblables à celles de Fessenheim et Bugey mais plus encore standardisées entre elles à quelques évolutions près) est maintenant prévu suivant un rythme moyen d'une unité tous les deux mois.

Avec un certain recouvrement avec ce palier de tranches, débute la réalisation d'un autre palier, composé de tranches de 1300 MWe de la même filière. Les deux premières d'entre elles (Paluel 1 et 2) ont obtenu leur autorisation de création ; des demandes ont été actuellement déposées pour quatre autres tranches identiques et pour plus d'une douzaine de tranches de 1300 MWe semblables aux précédentes, mais avec quelques modifications, apportées selon une organisation comparable à celle des 900 MWe. Toutes ces tranches sont équipées de chaudières du même constructeur, Framatome, et conçues par le même architecte industriel, Electricité de France.

Nous donnons ci-après à titre d'illustration concrête le programme de mises en services prévues pour les prochaines années :

.../...

---

(1) - Service Central de Sûreté des Installations Nucléaires.

Année de l'ordre d'exécution de la chaudière	Centrale	Année de mise en service	Puissance	Puissance totale correspondante
1974	Tricastin 1	1980	900	4 500 MWe
	Gravelines 1	1980	"	
	Dampierre 1	1980	"	
	Tricastin 2	1980	"	
1975	Gravelines 2	1980	900	5 400 MWe
	Dampierre 2	1981	"	
	Tricastin 3	1981	"	
	Gravelines 3	1981	"	
	Dampierre 3	1981	"	
	Tricastin 4	1981	"	
1976	Paluel 1	1984	1 300	5 800 MWe
	St Laurent B1	1981	900	
	Le Blayais 1	1982	"	
	Gravelines 4	1981	"	
	Dampierre 4	1982	"	
	St Laurent B2	1982	"	
1977	Le Blayais 2	1982	900	6 100 MWe
	Chinon B1	1982	"	
	Chinon B2	1982	"	
	Paluel 2	1984	1 300	
	Le Blayais 3	1983	"	
	Super Phénix	1984	1 200	
1978	Le Blayais 4	1983	900	4 000 MWe
	Cruas 1	1983	"	
	Cruas 2	1984	"	
	Paluel 3	1984	1 300	
1979	Cruas 3	1984	900	5 700 MWe
	Cruas 4	1985	"	
	Flamanville 1	1985	1 300	
	St Maurice			
	l'Exil 1	1985	"	
	Cattenom 1	1986	"	
Programme autorisé 1980 - 1981	Gravelines 5		900	11 800 MWe
	Gravelines 6		"	
	Paluel 4		1 300	
	St Maurice			
	l'Exil 2		"	
	Flamanville 2		"	
	Cattenom 2		"	
	Chinon B3		900	
	Belleville 1		1 300	
	Nogent s/Seine 1		"	
	Belleville 2		"	

Les difficultés de toutes sortes soulevées du point de vue de l'administration de la sûreté par la réalisation d'un tel programme n'ont pu être surmontées que grâce à des procédures bien adaptées et à la recherche de tous les avantages que peut offrir la standardisation. La standardisation présente aussi certains inconvénients, dont il importe de limiter autant que faire se peut les conséquences. Après un bref rappel du système réglementaire français et des principaux acteurs concernés, nous évoquerons longuement ce dernier aspect, sans doute le plus spécifique à la France.

2. LE SYSTEME REGLEMENTAIRE FRANCAIS : LES PROCEDURES ET L'INSPECTION

Le système réglementaire français repose sur un décret du 11 décembre 1963 modifié le 27 mars 1973.

Les principaux aspects de ce décret méritent d'être rappelés ici. Il fixe tout d'abord le régime d'autorisation des installations nucléaires de base, qui doivent faire l'objet d'une autorisation de création par décret du Premier ministre, puis sur le rapport du ministre de l'industrie, ministre chargé de la sûreté à qui le décret précité de 1963 donne également le pouvoir d'imposer toute mesure d'urgence sur une installation, y compris son arrêt immédiat.

Il est également prévu que l'autorisation de création ne peut être donnée qu'après enquête publique, avis d'une commission interministérielle compétente, et avis conforme du ministre chargé de la santé, qui dispose ainsi pratiquement d'une sorte de droit de véto.

L'autorisation de création marque l'accord de la puissance publique sur le projet envisagé par le futur exploitant : elle ne peut donc être donnée qu'après examen approfondi de la sûreté de l'installation concernée. Cet examen s'effectue sur la base d'un rapport préliminaire de sûreté présenté par l'exploitant à l'appui de sa demande d'autorisation.

L'autorisation de création explicite les prescriptions techniques jugées les plus importantes du point de vue de la sûreté. Si l'exploitant venait à souhaiter des modifications telles que ces prescriptions ne soient plus respectées, une nouvelle autorisation dans les mêmes formes lui serait nécessaire. Elle peut également imposer à l'exploitant la remise de dossiers d'étude ou de descriptions complémentaires dans des délais propres à permettre une intervention de l'administration suffisamment précoce pour éviter toute décision irréversible. Cette possibilité, essentielle lorsqu'il s'agit d'installations d'un type nouveau, est également utile pour le cas d'installations standardisées en permettant, nous le verrons, une meilleure ouverture au progrès technique. L'autorisation de création précise en outre qu'avant le premier démarrage de l'installation et l'approbation par le ministre de l'industrie de sa mise en service définitive, l'exploitant devra faire la preuve de la sûreté suffisante de ses installations dans un rapport de sûreté accompagné de règles générales d'exploitation. Il devra également apporter à celles-ci, toutes les modifications jugées nécessaires par le ministre.

.../...

Le décret précité de 1963, précise également les conditions dans lesquelles est effectuée la surveillance des installations nucléaires de base, en constituant notamment la fonction d'inspecteur des installations nucléaires de base ; ceux-ci, assermentés, disposent des pouvoirs d'investigations les plus larges tant chez les exploitants que par l'intermédiaire de ceux-ci, chez leurs sous-traitants dès lors qu'une demande d'autorisation de création a été déposée. Leur action se situe donc à toutes les phases de conception, de construction ou d'exploitation des installations.

Bien qu'ils travaillent pour l'administration, les inspecteurs des installations nucléaire de base jouissent de la plus grande liberté dans la conduite de leurs visites de surveillance : ils ont le droit de porter à leur rapport toutes remarques jugées par eux nécessaires même si a priori elles ne portent pas sur des systèmes importants pour la sûreté.

Si une tuyauterie peut inonder certaines parties de l'installation, cela concerne la sûreté, même si la tuyauterie appartient à un circuit tout à fait auxiliaire.

Le décret de 1963 modifié précise enfin que la réglementation technique est prise par arrêté du ministre de l'industrie et que, dans un délai précisé pour chaque autorisation de création, celui-ci doit voir soumis à son approbation la mise en service définitive de l'installation .

Pour conclure cette présentation générale, il convientde préciser pour mémoire, qu'à ce décret de 1963 spécifique au caractère nucléaire des installations et aux procédures qui en découlent, viennent s'ajouter d'autres textes plus généraux, notamment en matière de rejets d'effluents radioactifs liquides ou gazeux ou d'appareils à pression. Nous ne développerons pas davantage ici les caractéristiques de ces procédures.

En définitive, avec trois étapes importantes, la création qui intervient de façon très préliminaire, le démarrage qui marque le début du caractère véritablement nucléaire de l'installation et la mise en service définitive qui permet d'effectuer, en complément, un bilan du début d'exploitation, avec la remise par l'exploitant de dossiers complémentaires avant certaines décisions de construction importantes et irréversibles, avec enfin la surveillance des installations nucléaires de base et les demandes complémentaires qui peuvent à tout moment être faites par l'administration, le système réglementaire français permet -et implique- un suivi très détaillé de toutes les phases de l'existence d'une installation nucléaire de base.

Ce suivi est possible malgré l'ampleur du programme électronucléaire, grâce aux efforts déployés en permanence par l'administration et les experts travaillant pour son compte ainsi que grâce à une importante standardisation entre les tranches.

Nous examinerons successivement ces deux points :

.../...

3. LES ACTEURS CONCERNES

La sûreté c'est,en effet d'abord des responsabilités clairement définies aux mains de personnes capables de les assumer : ceci concerne aussi bien les exploitants que l'administration et ses appuis techniques.

L'exploitant est le responsable d'ensemble de son installation : en cas d'urgence, c'est lui qui peut intervenir le plus rapidement. La qualité de l'exploitant est la base même de la sûreté nucléaire. Il doit en avoir les moyens tant en ce qui concerne les capacités de réflexion qu'en ce qui concerne la formation du personnel. Quand on pense standardisation, on pense toujours unicité de conception ou de réalisation, car on n'attache généralement pas assez d'importance à l'exploitation. En réalité, par les possibilités qu'elle offre de mise en place d'équipes de réflexion importantes pour les cas d'urgence, de réalisation de simulateurs, de formation du personnel pratiquement en double commande pendant plusieurs mois sur des tranches déjà en exploitation, la standardisation au niveau de l'exploitation, même si l'on y pense moins souvent est certainement très importante. Electricité de France, unique exploitant des centrales du programme électronucléaire a entrepris, en liaison avec certaines demandes faites par l'administration, un certains nombre d'actions pour tenter d'utiliser au mieux les possibilités ainsi offertes.

Mais la clarté des responsabilités au sein de l'administration est également importante. Aussi le ministère de l'industrie a-t-il créé en 1973 le service central de sûreté des installations nucléaires, service aujourd'hui rattaché à la direction de la qualité et de la sécurité industrielles. Ce service est chargé de préparer et de mettre en oeuvre toutes actions techniques du département relatives à la sûreté nucléaire, d'examiner les programmes du commissariat à l'énergie atomique qui s'y rapportent de suivre les travaux de recherches, de recueillir toutes imformations utiles dans le domaine de la sûreté nucléaire. Il propose et organise l'information du public et d'une façon générale examine les mesures propres à assurer la sûreté des installations nucléaires.

Il a donc une mission très globale et en quelque sorte le rôle de responsable administratif d'ensemble dans le domaine de la sûreté nucléaire.

Il dispose pour l'accomplissement de cette mission d'appuis techniques importants au sein des directions interdépartementales de l'indutrie et de l'institut de protection et de sûreré nucléaire du commissariat à l'énergie atomique.

Des groupes permanents d'experts sont également placés auprès du chef du service central de sûreté des installations nucléaires ; certains inspecteurs des installations nucléaires de base ainsi que des membres d'autres ministères, particulièrement du ministère de la santé, participent en tant que de besoin aux réunions de ces groupes.

4. LA STANDARDISATION : AVANTAGES ET DIFFICULTES

Les avantages liés à la standardisation pour les exploitants et constructeurs sont bien connus ; l'intérêt et les difficultés qu'elle représente pour l'administration chargée de la sûreté le sont moins et l'expérience française est à cet égard importante.

DES ETUDES TRES POUSSEES

L'existence d'un grand nombre de tranches très semblables justifie tout d'abord la réalisation de certaines études ou recherches de caractère très appliqué, mais qui n'entrent pas dans le cadre des études jugées habituellement nécessaires pour la délivrance des autorisations. Ces études effectuées pour une tranche sont ensuite transposées à l'ensemble des tranches de la série. Quelques exemples méritent d'être cités : pour Fessenheim, ont été réalisées des études très complètes de fiabilité des systèmes qui permettent notamment d'apprécier plus complètement les conséquences de l'application du critère de défaillance unique utilisé lors de la conception et d'affiner les règles générales d'exploitation. Leur confrontation avec l'expérience d'exploitation sera également riche d'enseignements. De même ont été entreprises des études du comportement post-accidentel des installations allant au-delà des durées habituellement examinées. Enfin, méritent d'être citées les études de défaut complet de fonctionnement de certains systèmes, ATWS bien sûr, mais aussi perte totale des alimentations électriques tant externes qu'internes ou perte de toute source froide. A citer dans le même esprit, quoique sur un plan un peu différent, le bien fondé d'une qualification systématique plus poussée des matériels aux conditions accidentelles auxquels ils peuvent être soumis.

Les études de sûreté d'une tranche apportent également des améliorations pour la suite : ainsi à Fessenheim et à Bugey l'injection de sécurité était mise en service sur coïncidence des signaux basse pression et bas niveau pressuriseur. Lors de l'autorisation de démarrage de ces tranches, le problème de la non représentativité du niveau pressuriseur en cas de rupture en phase vapeur de celui-ci avait été étudié et il avait été montré que l'opérateur dispose d'un délai de l'ordre d'une demi-heure pour déclencher normalement l'injection de sécurité en pareil cas. Ce délai a été jugé suffisant, mais, bénéficiant de cette expérience, les tranches suivantes sont conçues pour que la mise en service de l'injection de sécurité intervienne sur seul signal basse pression primaire. A noter, sans parler ici de tous les enseignements tirés de Three Mile Island qui est l'objet d'une autre session, qu'à la suite de l'accident, le "Backfitting" des progrès réalisés ultérieurement en ce domaine a été décidé. Les travaux correspondants sont maintenant achevés.

EXPERIENCE D'EXPLOITATION ET REGLEMENTATION

La standardisation permet aussi l'acquisition exploitable et transposable d'expérience d'exploitation. Ainsi, l'expérience de démarrage des deux tranches de la centrale nucléaire de Fessenheim et des quatre tranches de 900 MWe de la centrale nucléaire du Bugey a fait apparaître un nombre d'incidents sur chaque site en forte diminution d'une tranche à la suivante alors que, du fait de différences de détail entre les centrales de Fessenheim et de Bugey, la première tranche de 900 MWe de Bugey, démarrée après la seconde de Fessenheim, a connu plus d'incidents que celle-ci.

Quand une difficulté apparaît, les moyens à mettre en oeuvre peuvent en outre être définis directement pour toutes les installations sans qu'il soit nécessaire d'élaborer un texte de portée générale dont les inconvénients et les avantages doivent être soigneusement pesés en tenant compte alors de tous les types d'installations existants.

A côté de ces avantages, diverses difficultés doivent être surmontées bein entendu, pour la sûreté comme pour les autres aspects de la construction et de l'exploitation des tranches nucléaires, tout problème rencontré sur une tranche peut se répercuter sur les autres et, quand une solution est définie pour une tranche en construction, elle doit être apportée aux autres tranches et adaptées de façon appropriée aux tranches en fonctionnement.

Des difficultés peuvent aussi être rencontrées dans l'adaptation du projet standard aux différents sites : les questions les plus délicates rencontrées en France sont sans doute celles du séisme ou de tenue de l'installation aux agressions externes, notamment explosion : l'exploitant se trouve contraint d'apporter des modifications au standard et ces modifications peuvent nécessiter des analyses plus complexes que celles qu'auraient nécessité un projet directement conçu pour le site envisagé, même si globalement le projet demeure aussi sûr et si l'exploitant y trouve des avantages économiques. C'est ainsi que les tranches nucléaires de Cruas ont dû être montées sur appuis antisismiques, appuis originaux en France qui ont nécissité, tant de la part des constructeurs que de la part de l'administration et de ses appuis techniques, un effort d'analyse important. A Gravelines, des précautions particulières ont dû être prises sur toutes les ouvertures pour faire face à une onde de surpression incidente de 200 mbar, enveloppe de celles qui pourraient résulter sur la centrale d'une explosion survenant dans l'environnement industriel.

CONSERVER L'OUVERTURE AU PROGRES TECHNIQUE

Mais la difficulté la plus importante à surmonter est sans doute celle qui résulte de la nécessité de trouver le meilleur équilibre entre la standardisation et l'ouverture au progrès technique. Les connaissances progressent de façon continue, alors que les standards sont définis à certaines étapes précises. Il faut alors apprécier, au cas par cas, l'intérêt d'une éventuelle modification par rapport aux inconvénients d'une perte de standardisation ou aux difficultés d'un éventuel "Backfitting".

De même, si la conception reste inchangée, les procédés de fabrication peuvent évoluer, encore que de tellesévolutions doivent toujours être soigneusement étudiées et pesées. Pour chaque évolution de ce type, tout demeure affaire d'étude, de qualification et en fin de compte du savoir faire des constructeurs et de surveillance par l'administration.

C'est cette volonté d'ouverture au progrès technique qui conduit l'administration à rechercher sans cesse la meilleure connaissance des installations et, à l'occasion de chaque autorisation, à approfondir les analyses de sûreté au lieu de se contenter de noter la similitude avec les tranches précédentes et de s'appuyer sur la jurisprudence créée.

C'est ainsi qu'il paraît possible de préserver l'ouverture au progrès technique tout en cherchant à tirer tous les bénéfices possibles de la standardisation.

STRUCTURE, QUALIFICATIONS AND TRAINING OF THE REGULATORY BODY STAFF IN FINLAND

Jukka Laaksonen
Institute of Radiation Protection
Helsinki, Finland

Abstract

A small country approach to the regulatory personnel problems is presented. Recruitment of a staff with a relatively small previous knowledge and experience has been the only option available. In order to rapidly gain necessary knowledge and to jointly cover all aspects of nuclear technology most staff members have specialized to a limited technical field. Practical experience is acquired by combining the responsibilities for review and assessment and for inspections in such a way that each staff member carries out both efforts in his special field.

1. Introduction

The presentation gives an example of the approach to the regulatory personnel problems in a small country which has started its considerable nuclear program with almost no previous nuclear experience.

During the past ten years the regulatory organization has found a satisfactory structure and working methods. Its present role in Finnish nuclear power program can be regarded as significant. The utilities as well as supplier organizations have clearly recognized the necessity to take regulatory actions and requirements into account. Also the public reliance as expressed by the statements in the press and on political scenes seems to be relatively good.

2. Development of organization

In Finland the nuclear power plant construction permits and operating licenses are issued by the Ministry of Trade and Industry as provided in the Atomic Energy Law of 1957. However, the Ministry has no staff of its own to do actual regulatory work but it makes the decisions on the basis of statements given by other organizations. The nuclear regulatory functions are in practice performed by the Institute of Radiation Protection (IRP). The responsibility for that work is given to IRP director for nuclear safety. The director has in his command the IRP department of reactor safety. The department was formed in 1968 and up to the year 1975 the number of staff was increased rapidly.

The progress of Finnish nuclear projects and the regulatory staff increase is shown in figure 1. The present organization was established in 1975 and since then the staff has been almost unchanged. The size of staff is thought to be appropriate for present needs.

Several other governmental institutes are used as IRP consultants in special questions. In order to be able to give tasks to those consultants and to assess the work results the IRP has acquired also internal expert knowledge in corresponding fields.

3. Structure of present organization

The organization scheme of the Department of Reactor Safety is presented in figure 2. Also the number of technical staff in each operational unit is presented in the figure.

Groups responsible for mechanical components (MC), systems engineering (SE) and radiation protection (RP) have worked as separate operational units from the very beginning. Groups responsible for nuclear materials (NM) and buildings and structures (BS) were formed in 1975.

The review areas which are on the responsibility of each group are presented in table I. There are also certain boundary

areas which call for a close co-operation between groups. Especially can be mentioned reactor core design (SE-group, NM-group), fire protection (SE, BS), auxiliary systems (SE, RP) and design requirements for components and structures (all groups).

As can be seen the main organizational principle is to have specialists in all review areas. This enables an in-depth review in plant details but on the other hand most of the staff has only a limited knowledge about nuclear power plant as a whole.

Another lead idea in the organization is to use same people for both review and inspection efforts. Thus most of the regulatory officers visit quite often the plant sites and the shops of component manufacturers.

4. Staff qualifications

During the years 1969-75 when the regulatory organization was developed the growth of nuclear power program in Finland was rapid. Before that growth period only one small research reactor had been in operation since 1962 and the amount of engineers with nuclear experience was very limited. In the early years of nuclear power program the significant role of the regulatory body was not generally recognized. Thus the work within utility organizations and research work were found more attractive than regulatory work. IRP had difficulties in getting experienced engineers and no strict requirements on previous experience could be set.

The initial staff formed by four senior officers had mainly research background with the average professional experience of about ten years. From the staff recruited later on about one half came directly from universities or technical colleges after having completed their studies. Main part of the rest had industrial or research experience from one to four years and only five persons had been in their profession for more than five years.

The lacking experience has somewhat been counterbalanced by the opportunity to recruit talented engineers who had passed their university examinations with credit. Those young people have taken their job like a pioneering task and the good spirit inside the institute has inspired the people to often work overtime to improve their knowledge and to make all their duties. The positive attitude to the work is shown also by the fact that so far only a few people have left for another job.

In the early years of the nuclear power construction the limited knowledge, also on utility side, was taken into account by adopting a cautious attitude to difficult problems or doubtful solutions. Thus a conservative design with safety margins or multiple redundances was often preferred. Of course, this kind of approach does not help to identify the potential risks in a certain plant design and does not guarantee the safety of the plant as a whole.

5. Training of personnel

The inadequate previous experience has put special requirements on training of personnel. No formal training program has, however, been established and the number of internal training occasions has been small.

Instead of an own training program the staff participation in courses and seminars held by other organizations in Finland and abroad is encouraged and the costs are paid for. Most important nuclear or radiological safety courses which have been attended several times are organized by NRC and MIT in United States, Harwell in United Kingdom and Karlsruhe in Federal Republic of Germany. The written lecture material delivered at the courses is available to all IRP personnel and it is used in self-study. An important role in training have also played courses arranged by a national center for supplementary education of engineers. The length of such courses is 2 to 3 day and the scope is limited to some special subject, either nuclear or non-nuclear.

A general introduction to various aspects of nuclear technology is given with the help of a self-study package consisting of audio-visual training material. The package is produced in Sweden by a company which takes care of operator training for Swedish nuclear power plants. The same material is used by all Swedish and Finnish utilities in operator training.

The training in special areas has been on-the-job training instructed by more experienced colleques or self-study of applicable technical documentation. The technical documentation available to IRP staff is quite exhaustive. It includes, for example, a collection of U.S. and German criteria and guides as well as industrial standards from many countries. Besides these we have found the DOCKET-collection published in USA very useful for self-study. For example many docketed FSAR's and the correspondence related to these are studied as reference material during licensing reviews. As a training tool can also be regarded the Standard Review Plans published in USA. That documentation came just on time for the operating license review of our first nuclear unit.

Not underestimating the possibilities of organized training I think, however, that the most effective means in acquiring in-depth understanding is the combined duty as reviewer and inspector.

When a young regulatory officer participates the licensing review for the first time his work is necessarily based on guidance given in Standard Review Plans or other literature. If he then is to review an almost identical plant he hardly can do it much better but he only repeats most things he has already done before. But if he in between has surveyed construction activities or commissioning and operation on site he can base the second review on his practical observations and engineering judgement. This

conclusion has come very clear during operating license reviews of the second units of our twin plants. In fact, many of the weak points in our plants have been uncovered during these repetition reviews and this has led to certain backfitting in the units already in operation.

Another instructing job combination is given to those officers reviewing the accident analysis. They are also responsible for having made comparative accident analysis with the help of an independent research institute. That work forces them to get profound knowledge about calculational methods and plant features having influence on results.

As to the training in future we see the need to give a broader scope of all nuclear matters to the whole staff having so far specialized in their own problems.

TABLE I

SPECIAL FIELDS AND RESPONSIBILITIES OF GROUPS IN THE DEPARTMENT OF REACTOR SAFETY

Mechanical components group

Component dimensioning
Stress analysis
Material questions
Fabrication and installation technology
Component quality control and quality assurance

Systems engineering group

Reactor physics
Thermal and hydraulics
System analysis and testing (most plant systems)
Instrumentation and control
Electric power
Operator licensing

Buildings and structures group

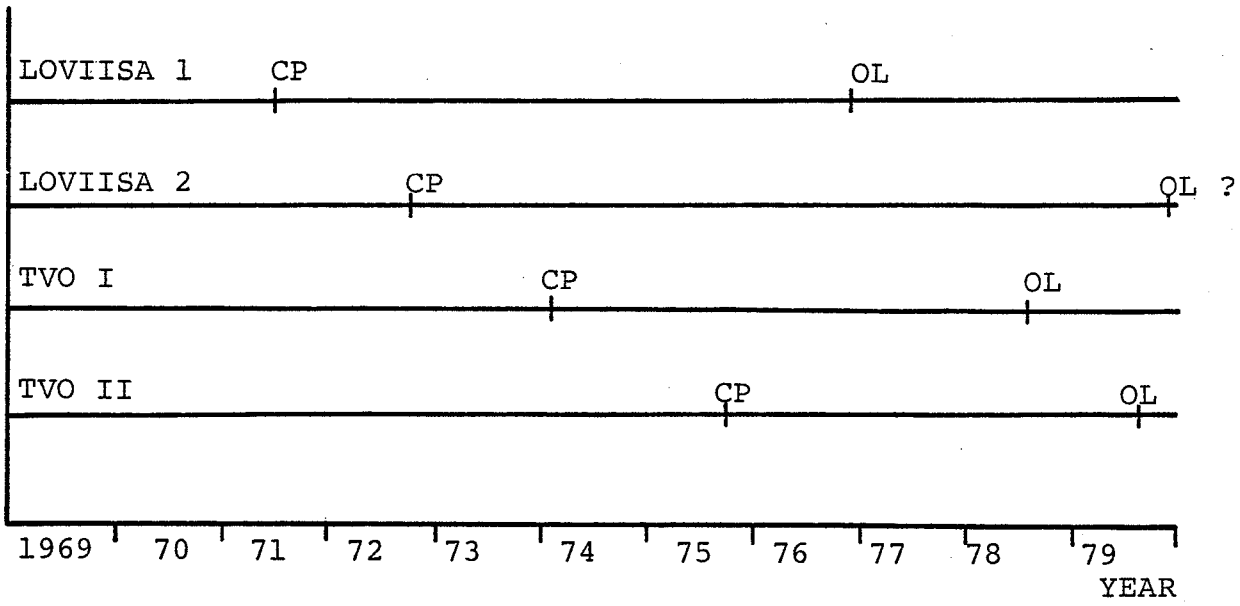
Construction technology
Structural design
Fire protection

Radiation protection group

Radiological protection and monitoring
Waste management
System analysis and testing (drainage, leakage collection, water purification, ventilation systems)
Containment leak-tightness
Emergency plans

Nuclear materials group

Fuel design, fabrication and quality assurance
Fuel transport, handling and storage
Safeguards of nuclear materials
Industrial security



CP = CONSTRUCTION PERMIT, OL = OPERATING LICENCE

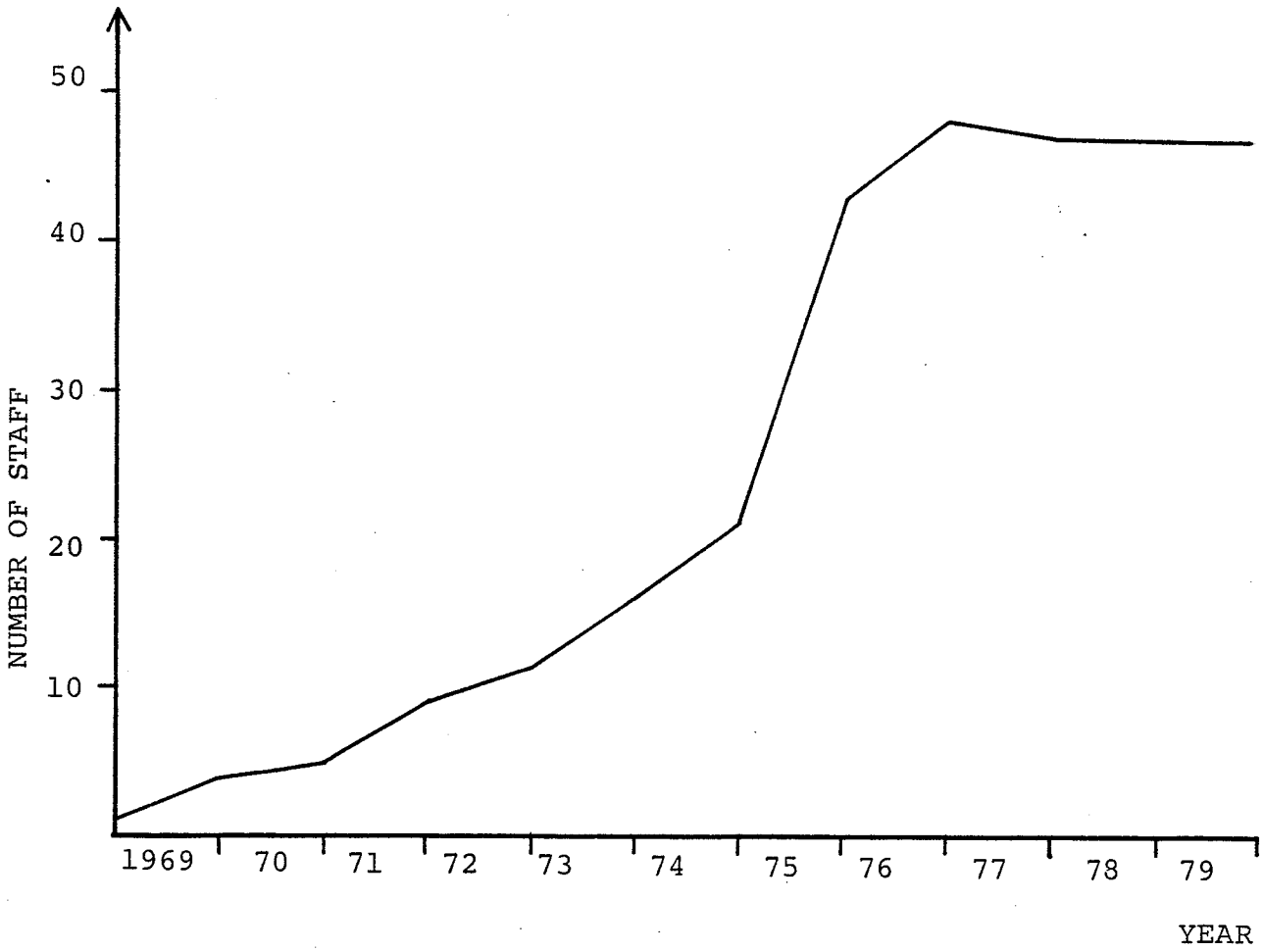


FIGURE 1. PROGRESS OF NUCLEAR POWER PLANT PROJECTS AND NUMBER OF REGULATORY STAFF

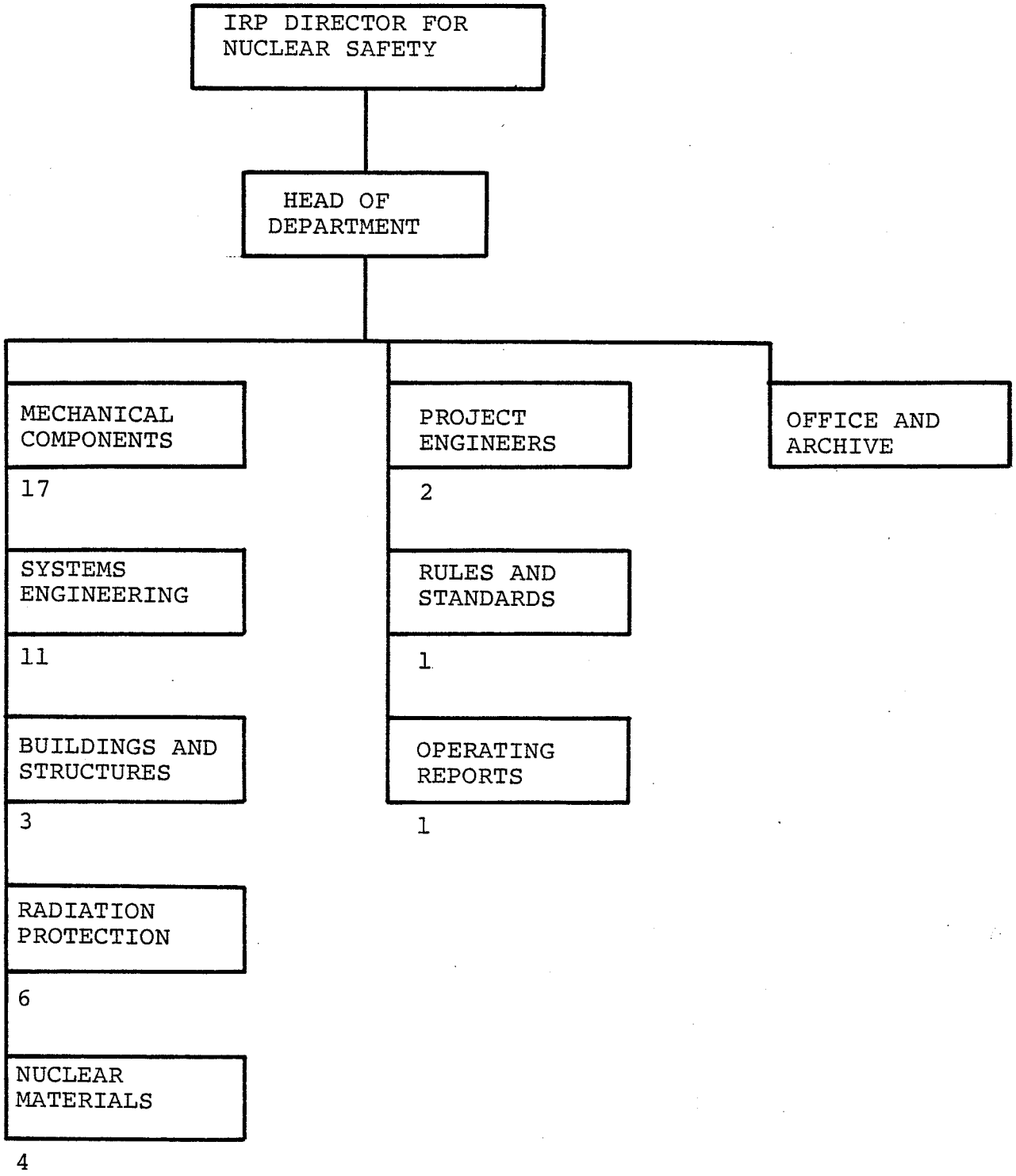


FIGURE 2. ORGANIZATION OF THE DEPARTMENT OF REACTOR SAFETY

REGULATIONS AND THE LICENSING PROCESS IN AUSTRIA

Dr. Herbert U. Matulla
Österreichische Studiengesellschaft für
Atomenergie
Vienna - Austria

Eine Rückschau über das Genehmigungsverfahren, das von 1971 bis 1978 dauerte, zeigt auf, welche Gesetze, Vorschriften und technische Regeln bei der Beurteilung der Sicherheitsaspekte des Kernkraftwerkes herangezogen wurden und durch welche Organisationen dies erfolgte. Die interne Organisation des österreichischen Hauptsachverständigen wird skizziert. Beispiele der Detailarbeit werden erläutert. Die Wichtigkeit der intensiven Zusammenarbeit der Fachgruppen und das Problem der gleichmäßigen Prüftiefe werden betont.

A review of the licensing process which took place from 1971 to 1978 shows which laws, regulations and standards were used in checking the safety aspects of the nuclear power plant and which organisations participated in the licensing process. The internal organisation of the Austrian main-expert in the procedure is illustrated. Examples of detail-work are explained. The importance of intensive co-operation of the different technical groups and the problems of comparable examination depth are underlined.

Exactly one year ago - it was the 5th of november 1978 - the Austrian referendum on nuclear power brought the result of 50,5% NO to 49,5% YES.

As a consequence of this referendum a law passed the parliament in December 1978 in which the industrial use of nuclear energy for production of electricity is prohibited in Austria.

These facts stopped the Austrian licensing procedures which began in 1971. The Nuclear Power Plant Zwentendorf was nearly completed and would have been able to start nuclear tests in the 2nd quarter of this year.

1. Data of the Nuclear Power Plant

Allow me to give you a short impression of the Nuclear Power Plant GKT (Gemeinschaftskernkraftwerk Tullnerfeld in Zwentendorf) (Pict. 1 and 2).

Today the disassembling and the conservation of the most systems is finished. Some systems (as air conditioning, fire protection ...) are still in operation.

2. Legal aspects

According to the Austrian laws construction and operation of a Nuclear Power Plant requires different official procedures, which are not necessarily connected to each other.

The picture (Pict. 3) gives you an overview on these procedures, the most important and time consuming of them running at the Ministry of Health and Environmental Protection. I will describe it in short words.

After the decision of the country owned electricity utilities to built a Nuclear Power Plant in 1971, the licensee first handed over the primary safety assessment report to the Ministry of Health and Environmental Protection.

This authority made the decision to appoint two major organisations (SGAE and TÜV) and some single persons as experts for the starting procedures to examine the papers, calculations and programs and to formulate conditions for the construction permits and finally for the operating licence.

This procedure is schematically shown in the next picture (Pict. 4).

Up to 1978 approximately 46 construction permits where handed over to the licensee. The first of three planned operation licences was in preparation.

Now let us have a short look on the main expert organisation which where active in this procedures (Pict. 5).

And now a closer look to my company (Pict. 6).

The Austrian Studiengesellschaft für Atomenergie is a private organisation (Ges.m.b.H.) but the state owns 51%.

It was founded in 1957. Its main location is in Lower Austria (Seibersdorf), where we have a TRIGA Reactor and approximately 450 employees.

The licensing group was (and is) located in Vienna, because of the contacts to Gouvernment and so on.

As you can see the licensing group is organised in a traditional way. Its development of workforce during the procedure is given in the following picture (Pict. 7).

3. Examples of our licensing work

Example 1:

In the first example I will give you an impression of our work with the High Pressure Core Injection System (Pict. 8).

As there were no Austrian regulations we used the German RSK-Guides for boiling water reactors (1974) and the US-Regulatory Guides.

The following pictures show the important points out of this rules (from the standpoint of system analysis) for this system (Pict. 9 and 10).

Analysis of the HPCI-System showed, that (from the standpoint of specification and regulations) the tests would be successful.

But when we examinated the case of a well startet HPCI System (with normal power) and postulated a loss of normal power in this situation the switch over to emergency power would happen automatically and everything seemed to be o.k.

Except the fact, that this system, which looks so independent (Curtis turbine, driven by steam from the reactor vessel) needs an oil supply system for lubrication and control.

And during the starting operation of the diesels the oil system would have lost the pumps,

- and the turbine
- and the high-pressure pump their lubrication.

The simple action of feeding at least one of the two pump-motors for oil supply with battery-powered electricity solved this problem.

Example 2:

Cable inspection.

After the Browns Ferry fire the authorities demanded first a 10% inspection of safety related cables to guarantee cleanness of redundancy.

When one violation was detected the inspection was extended to 100%.

Some violations or small distance points between redundancies were found. Separation was made by fire protection means or by removing to another rack.

For the so called "zero redundant" cables which where allowed to change between redundancies, careful calculation and dimensioning of fuses were made.

To master the problem of inspecting some 1000-cables (of nearly the same colour) along their ways on the racks an electronic device was developed which allowed us to mark cables (during operation) on their shield with a signal and to follow this signal with a receiver.

So the inspection and control of documentation of cableways was done within 3 month by 10 men.

The discovered violations or problematic spaces where discussed in the so called "fire protection group" and corrective measures where taken.

As a wish-dream remained the idea of cables of different colours for the corresponding redundancies.

Example 3:

Second failure tables.

When preparing the operation licence the question arose how to handle malfunction or failures in very different fields.

The leading idea was, that there has to be always the necessary amount of standby-safety system capacity.

But that standby capacaty in worthless or of minor worth if the ability of detecting failures and of starting safety actions and of doing this actions is reduced.

We have the strong feeling that from our basic philosophy of the plant every "first single failure" would do no harm.

But if there are more actual failures (malfunction, repairs etc) at a time period an analysis has to show that there is no unallowed influence on the described abilities of the safety systems.

The results of these analysis were our 2nd failure tables, which combined the system side (system function) with the electronic and mechanic side of the detection systems (Pict.11).

We devided the results in the following groups

a) no special counteractions (beside repair) necessary

b) longer repair time (~ 100 h)

c) short repair time (10 h)

d) immediate shutdown of the plant.

These tables where developed on request of ÖSGAE by KWU.

Although we cannot use these tables now, ÖSGAE has a program with the OECD-Halden-project to develop from these tables a computer aided tool to give operators a help for decissions.

4. Conclusions

I hope I was able to give you a compressed survey of the Austrian licensing procedure and how the work was organised and done.

When we look back we can say that the Austrian procedure worked - more or less - effective. We think that it would have resulted in the operations licence in 1979.

The fact that there were a lot of parallel procedures (of minor importance) may bring for example the danger that at one side things are requested which do not fit the needs of the other side. But on the basis of our laws we had to live with this disadvantage. As the same experts worked sometimes for different authorities this danger was reduced.

From the standpoint of my own organisation we learned that the traditional groups as

- mechanical engineering
- electrical engineering
- electronics
- chemical engineering
- physic
- health physics

have to hold always an intensive contact to make sure that problems are seen in their correct size and with all their effects going over the group borders.

It is a recommendation on licensing organisations too, to have an eye on the company organisation of the constructor and to check fields I would call "border fields" between constructor groups.

And as a reminder for self-control, such a group has to try always to see the depth of examination from the standpoint of safety. This is a demand, not easy to fulfil, but one has to do its best (Pict. 12).

5. Todays situation in Austria

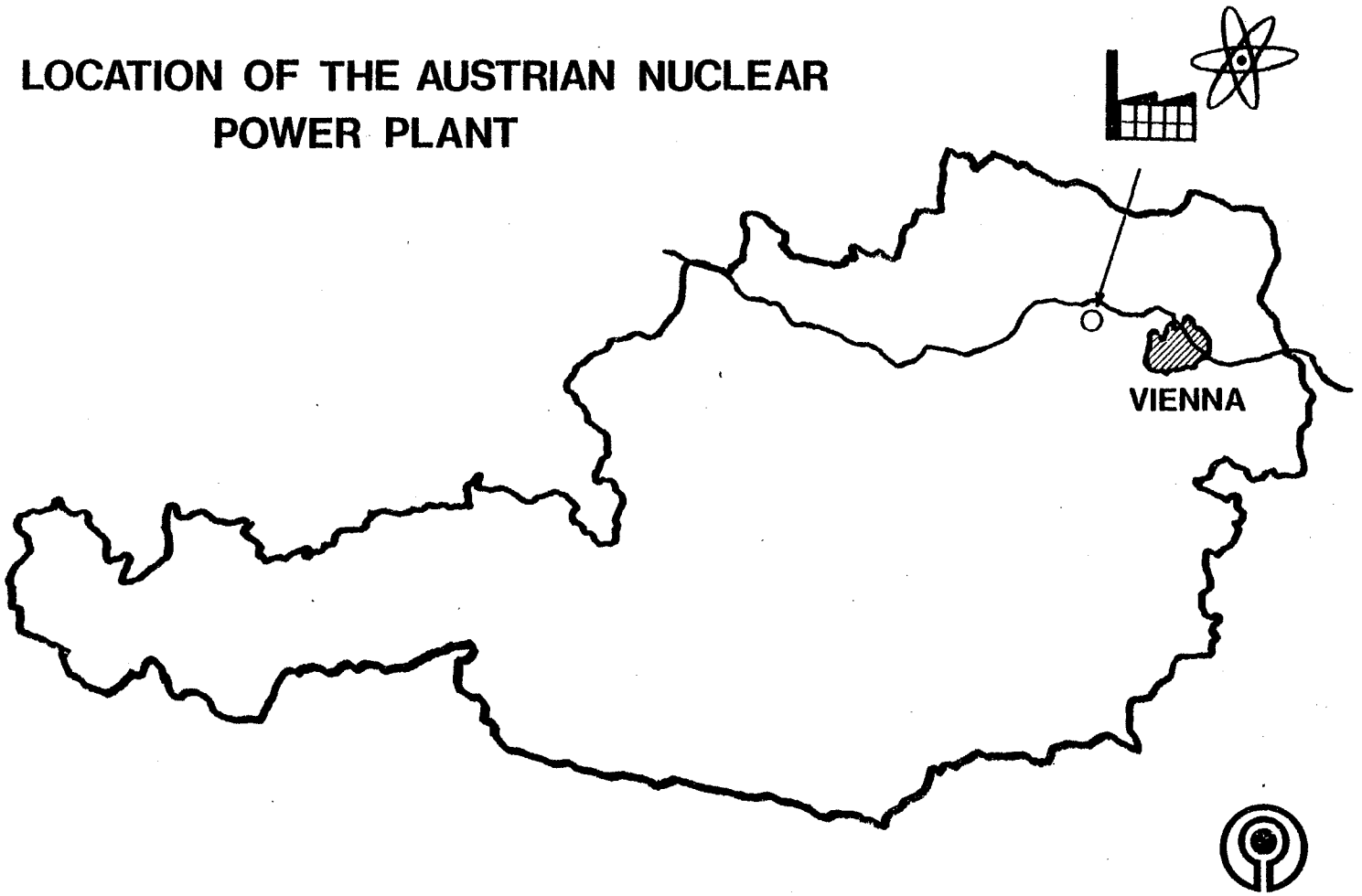
Analysis of the work we did - and of the work we hope, we will have to do in the future - is going on and we believe, that this future is not too far.

There are also strong interests in Austria in favour of taking Zwentendorf in operation.

But it is (you know) a political and not a technical problem to do this.

From the pure technical standpoint it would take at least one year to reassemble and retest the systems for startup clearance.

I thank you for your attention. If there are questions, I will try to answer them.



Picture 1

GKT - NUCLEAR POWER PLANT

BOILING WATER REACTOR

CONSTRUCTOR	:	KWU / AEG
OWNER	:	GEMEINSCHAFTSKRAFTWERK TULLNERFELD GMBH
THERMAL POWER	:	2100 MW
ELECTRIC POWER	:	730/700 MW
FUEL ELEMENTS	:	484
CONTROL RODS	:	113
STEAM PRESSURE	:	72 bar
SPEED OF TURBINE	:	3000 rev/min
CONDENSER COOLING	:	DANUBE WATER

Picture 2

AUTHORITIES FOR THE LICENSING OF THE AUSTRIAN NPP

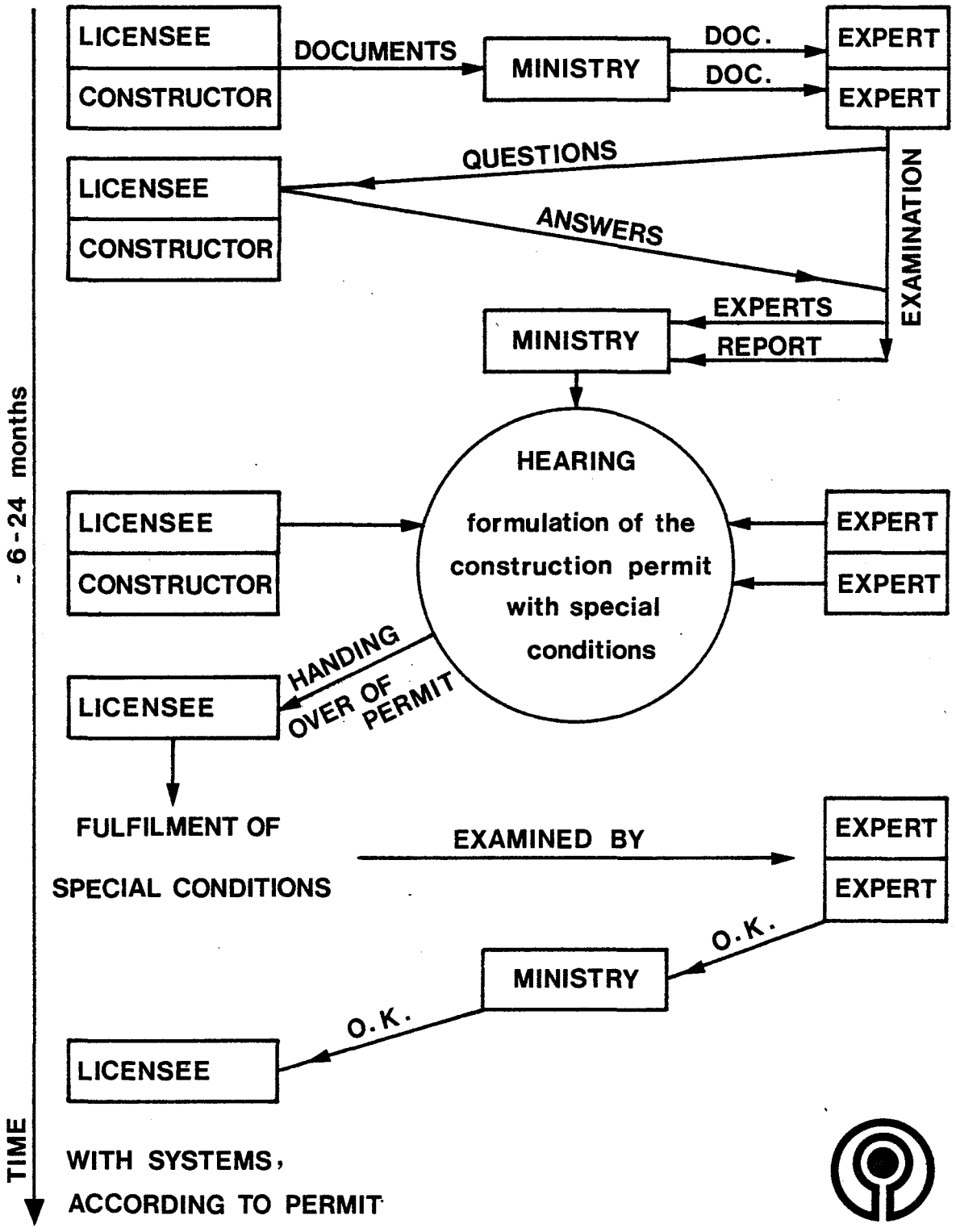
AUTHORITY	RESPONSIBLE FOR
Ministry for Health and Radiation Protection	radiation protection law, -regulations construction permits for safety related systems, operation permit
Ministry of Civil Engineering and Technology	steam - pressure - vessels civil engineering and fire protection
Ministry for Agriculture	water laws, clearing the site from the wood
Ministry for Commerce, Trade and Industry	laws of electrical engineering high tension lines - laws

Picture 3/1

AUTHORITY	RESPONSIBLE FOR
Office for Aerial Navigation	Location and Height of Stack
Ministry for the Interior	Plant Protection
Ministry for Foreign Affairs	IAEA - Contacts, Regulations
Office for Labour Inspection	Protective Labour Legislation

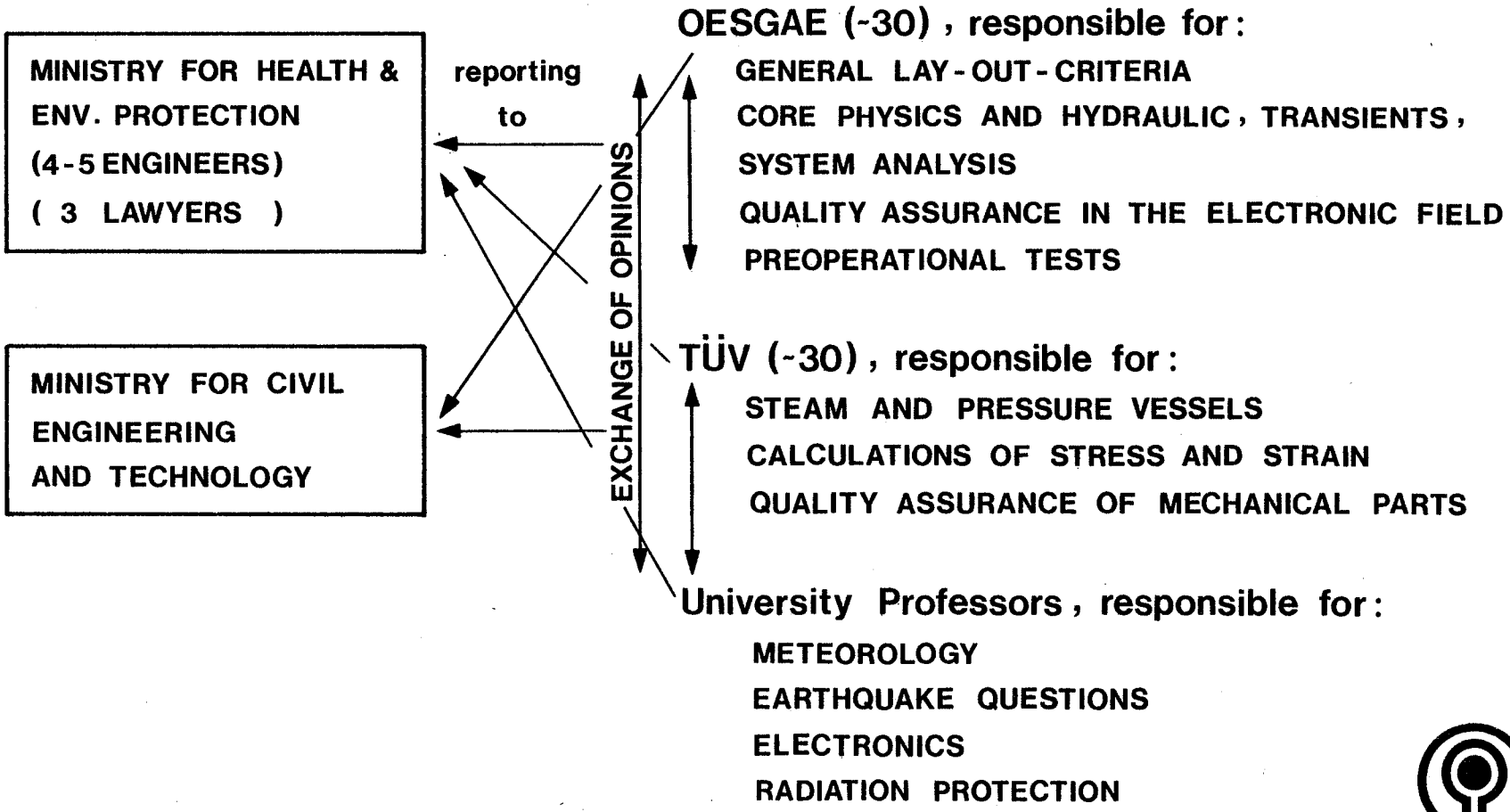
Picture 3/2

THE GENESIS OF A CONSTRUCTION PERMIT



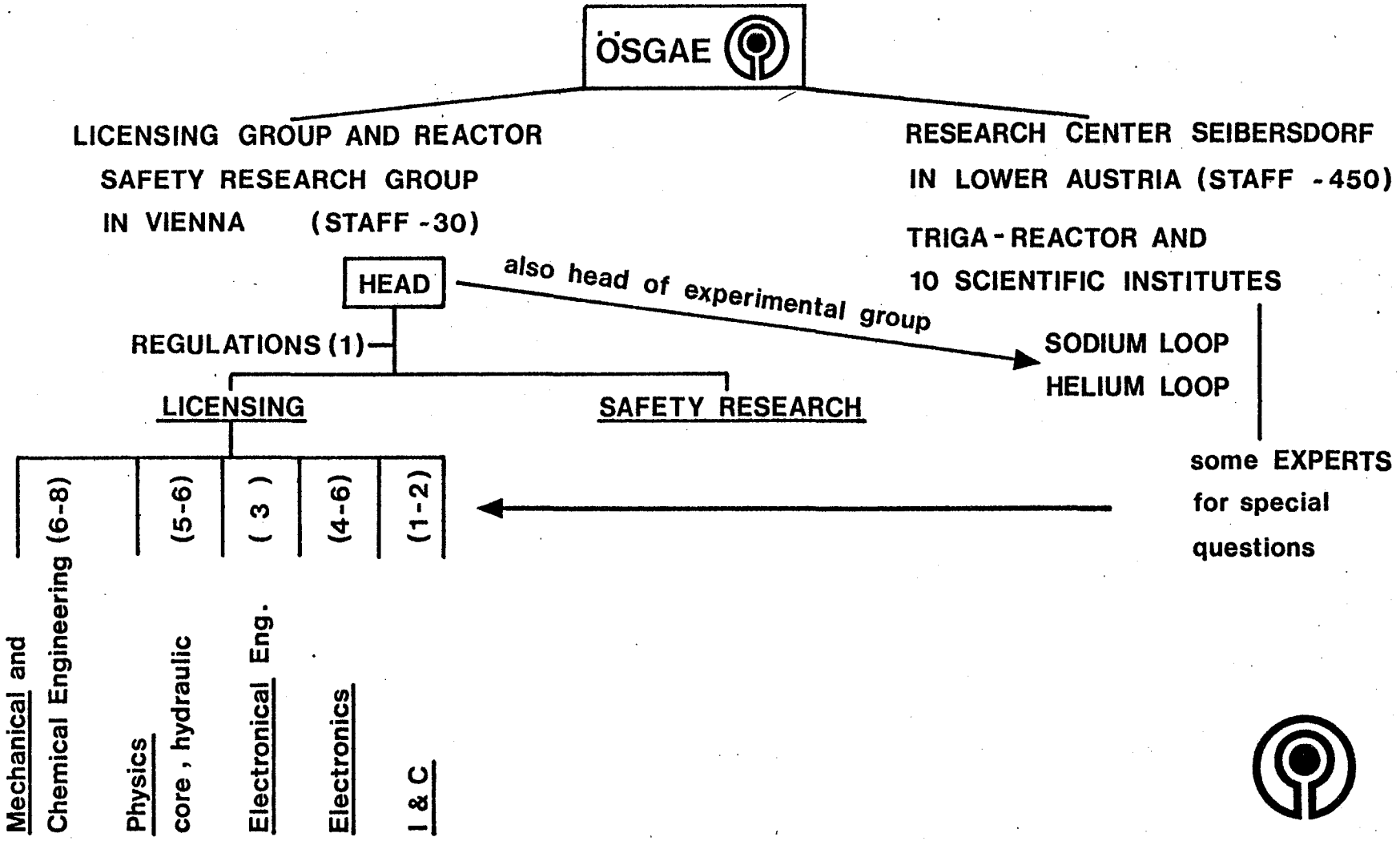
Picture 4

COOPERATION OF THE MAIN MINISTRIES AND APPOINTED EXPERTS



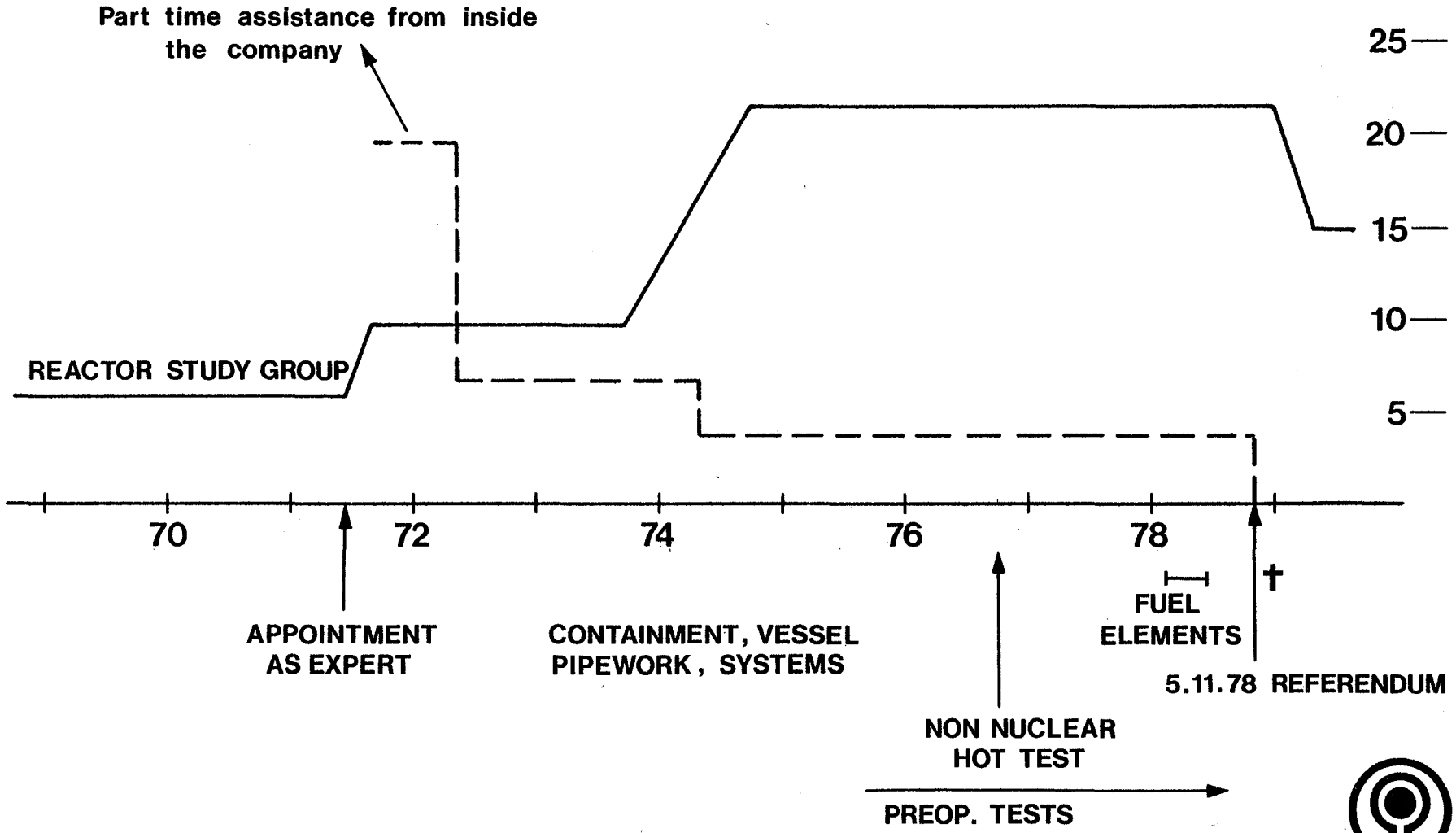
Picture 5

ORGANISATION OF ÖSGAE AND ITS EXPERT GROUP



Picture 6

WORKFORCE OF THE LICENSING GROUP VERSUS TIME



Picture 7

HPCI - SYSTEM

STEAM DRIVEN CURTIS TURBINE

OPERATION RANGE: 10 - 90 bar

BETWEEN LEVELS DEEP 1

HIGH

INITIATED BY:

CONTAINMENT PRESSURE >1,25 bar 100%

LEVEL < DEEP 1 50%

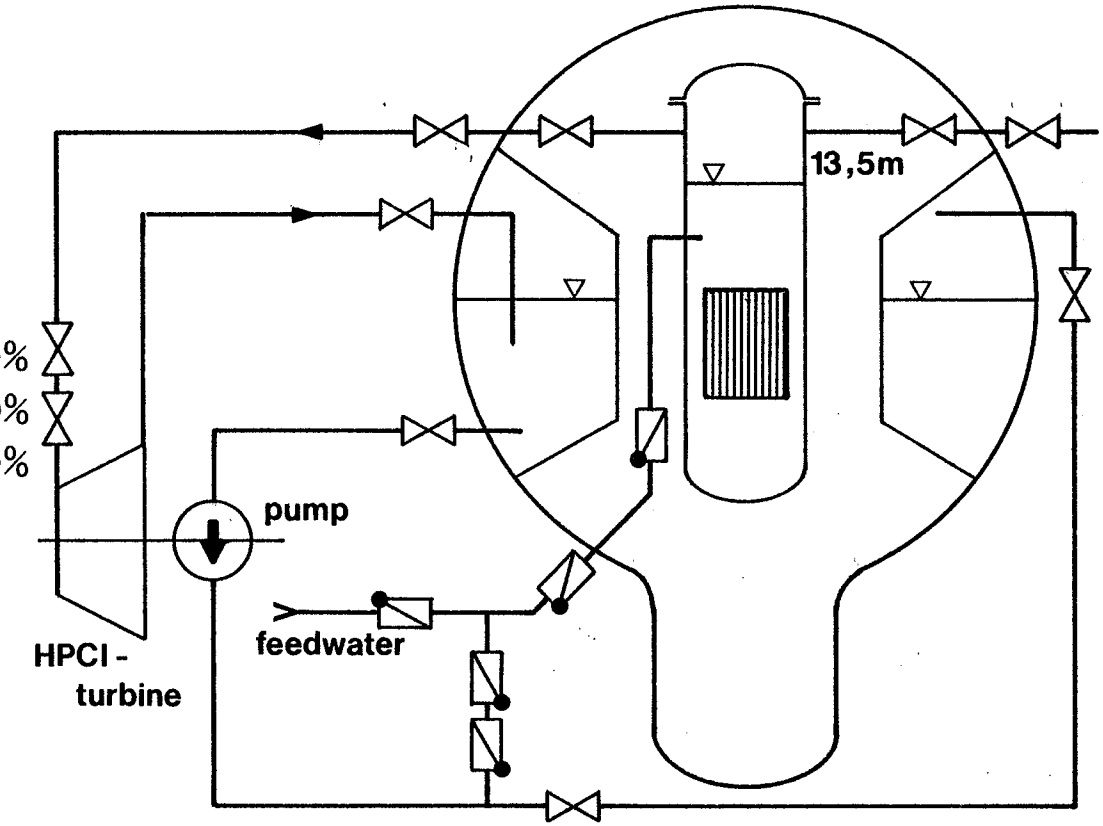
LEVEL < DEEP 2 100%

SWITCHED OFF BY:

LEVEL EXTREME HIGH

TURBINE PROTECTION

REACTOR PRESSURE < 12 bar



Picture 8

GERMAN RSK-DIRECTIONS FOR BWR DRAFT E 7.74

4. CORE COOLING SYSTEMS, PRIMARY BOUNDARIES
(DEFINITIONS, MATERIAL, MANUFACTURING, QA)

4.1.4 OPERATION
(INSPECTIONS, PRESSURE TESTS, ACCEPTABLE FLAWS, ...)

4.3 PREDICTIONS FOR ACCIDENT ANALYSIS

4.5 CORE COOLING AFTER ACCIDENTS
(THERE HAVE TO BE COOLING SYSTEMS WHICH ARE ABLE TO HOLD FUEL ROD TEMP. < 1200 °C, ... FLOW RATES, REDUNDANT ENERGY SUPPLY, ...MORE ORIENTED ON LOW PRESSURE CORE COOLING)

5. INSTRUMENTATION

7. EMERGENCY POWER SUPPLY
(DIESEL GENERATORS FOR IMPORTANT SYSTEMS, 2nd POWER SUPPLY, BATTERIES....)

Picture 9

USNRC REGULATORY GUIDE 1.68 (1973)

INITIAL STARTUP TEST PROGRAM TO DEMONSTRATE REMOTE SHUTDOWN CAPABILITY FOR WATER COOLED NPP

APPENDIX A

1. PREOPERATIONAL TESTING

Preoperational tests should demonstrate that structures, systems and components will operate in accordance with design in all operating modes and throughout the full design operating range.

→ this led to the non nuclear hot test

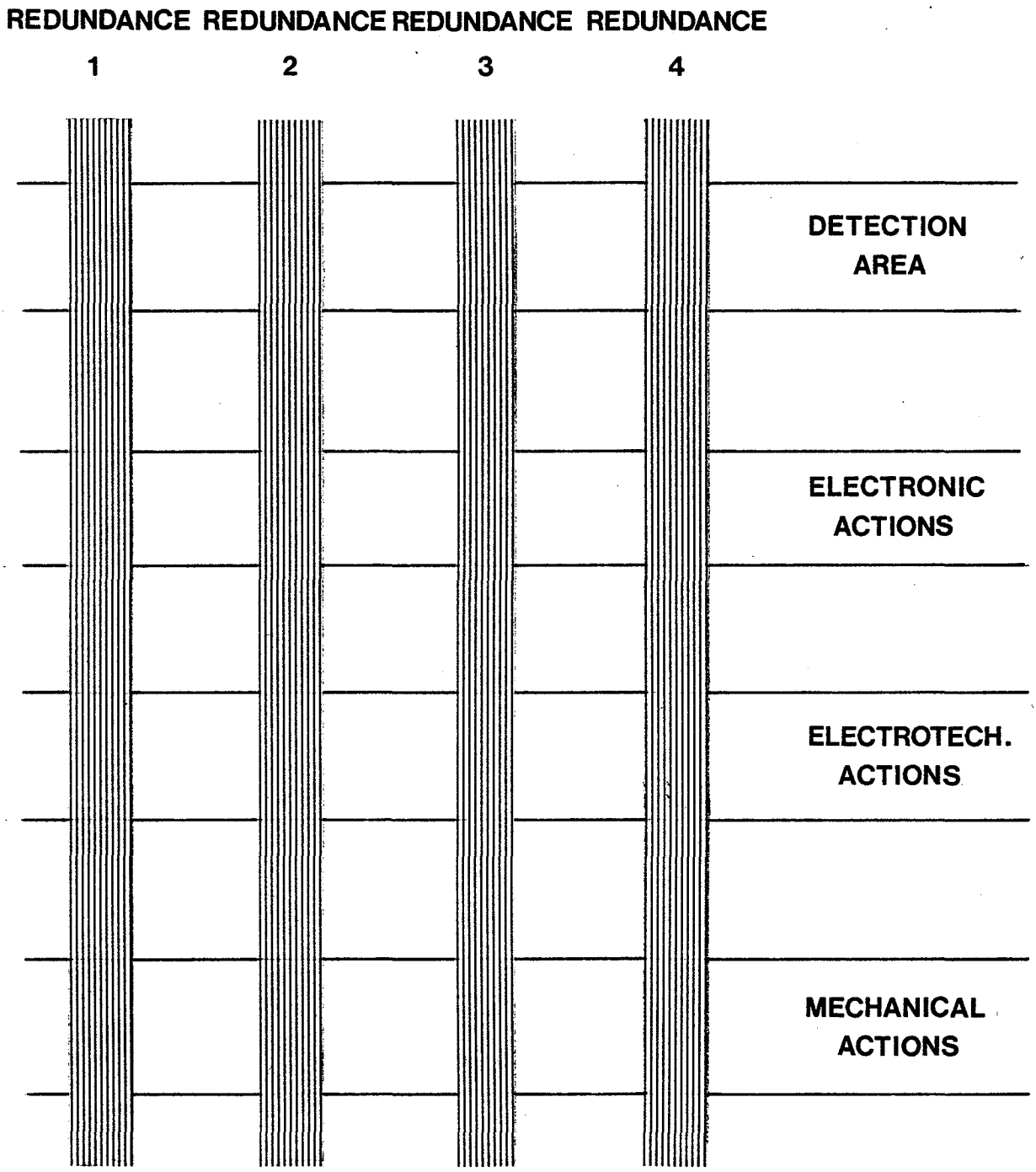
h. ENGINEERED SAFETY FEATURES

(1) ECC – SYSTEMS

(b) DEMONSTRATE OPERABILITY USING NORMAL AND EMERGENCY POWER SUPPLIES

→ here we also analysed the system behaviour during the switching over time

Picture 10



FAILURES IN DIFFERENT AREAS HAVE TO BE CHECKED UNDER THE ASPECT OF SUFFICIENT CAPACITY OF SAFETY SYSTEMS

Picture 11

REGULATORY ORGANIZATION

OF TURKEY

by N. Aybers

1. INTRODUCTION

The main concern of a developing country is that, there is at certain administrative levels a lack of proper appreciation of the size of the problem and the amount of effort needed to maintain safety standards and surveillance over the nuclear power project.

In addition, it can be assumed that in the early stage of a nuclear power development programme, resources available for staffing and organizing a regulatory body are limited. In general, in developing countries, the regulatory bodies have no staff allocated for the nuclear power plant, or a small number of staff members having little experience.

It should be noted also that during the initial period of the regulatory body, National Safety Codes of Practice and Safety Guides for nuclear power plants are not available. Therefore, it can be assumed that the steps required to ensure the maintenance of safety standards may cause serious difficulties.

2. LEGAL REGIME APPLICABLE

2.1. Competent Authorities

Act No.6821 of 27 th August 1956 for The Establishment of an Atomic Energy Commission amended by Act No.7190 of 14 th January 1959 made Turkish Atomic Energy Commission (TAEK) responsible for protection of public health and safety, and national security in the peaceful uses of atomic energy.

./..

Act No.7256 of 3 th April 1959 amended by Act No.234 of 5 th January 1961 laid down the way of the application of the Atomic Energy Programme of Turkey.

The TAEK is responsible for defining safety in respect of all nuclear activities and for drawing up regulations concerning radiation protection and the safe utilisation of nuclear installations. The TAEK which is the ruling body for licensing of nuclear installations, is attached to the Prime Minister's office and is chaired by a Minister of State appoint by the Prime Minister.

The first legislation in Turkey relating to protection against ionizing radiations was an Act adopted in 1937 and covering the regime applying to X-ray appliances used for medical purposes. However the scope of this Act was too limited to provide the regulatory framework necessary for developing the uses of ionizing radiations.

Decree No.6/7946 of 25 th April 1967 of the Council of Ministers laid down general provision applying to protection against ionizing radiations. This Decree was subsequently completed by detailed procedures on safety measures against ionizing radiations. TAEK Regulations of 16 th December 1968 laid down rules for the safe utilisation of ionizing radiations.

The special conditions governing the granting of authorisations to the nuclear installations are specified in the Decree No.7/9141 of 6 January 1974 of the Council of Ministers.

A Committee on Reactor Safeguards was established by the Decree No.7/9141 and the members are appointed by the TAEK for 3 years term. This Committee is responsible to TAEK. The Committee studies reactor safety and advise the TAEK on the safety aspects of the applications for licenses for nuclear installations. It is also responsible for the preparations of Safety Regulations issued by the TAEK.

./..

2.2. Licensing Procedure

In Turkey, a three-step licensing procedure is applied for nuclear installations;

- Site approval
- Construction permit
- Operating license

However, the licensing procedures for nuclear installation other than nuclear power plant and for nuclear power plant are somewhat different.

The procedures for granting an operating license for nuclear power plant and reactor includes three sub-step.

- Fuel loading permit
- Reactor start-up permit
- Full load operation permit

In advance of fuel loading, TAEK must determine that all the commissioning tests are properly performed and, before reactor start-up must determine that start-up tests are completed, operating procedures are satisfactory and that the persons who are to manipulate the controls of the reactor are qualified.

Fig.1 provides a General Schematic for the better understanding of Nuclear Power Plant licensing process of Turkey.

In its responsibility to assure the health and safety of the public, the TAEK should establishes a general legal framework. Therefore the Atomic Energy Acts should be supplemented by Regulations, Codes, Standards and Guides which deal with the special conditions governing the granting of authorisations.

./..

2.3. Nuclear Safety Standards, Codes and Guides

The Agency has prepared Safety Codes of Practice and Safety Guides for nuclear power plants which are now available in published or draft form. Although these documents establish an essential basis and guidance for safety, they may not in some special situations always be entirely sufficient or entirely applicable (1).

The question whether these requirements would become a legal obligation may pose some difficulties. Therefore, a special bilateral agreement has been signed between TAEK and National Utility TEK on November 20, 1978 and the following Codes of Practice has been turn-out to TEK in english for a tentative application as national codes:

- Code of Practice on Safety in Nuclear Power Plants Siting;
- Code of Practice on Design for Safety of Nuclear Power Plants;
- Code of Practice on Safety in Nuclear Power Plant Operation;
- Code of Practice on Quality Assurance for Safety in Nuclear Power Plants.

In the same basis the following IAEA Safety Guides are under review for releasing as national guides:

50-SG-G2 Information to be submitted in support of licensing applications for nuclear power plants (Published).

50-SG-G3 Conduct of regulatory review and assessment during the licensing process for nuclear power plants.

./..

50-SG-G4 Inspection and enforcement by the regulatory body for nuclear power plants.

50-SG-S1 Earthquake and associated topics in relation to nuclear power plants.

50-SG-S2 Seismic analysis and testing of nuclear power plants.

50-SG-S3 Atmospheric dispersion in relation to nuclear power plant siting.

50-SG-S4 Site selection and evaluation for nuclear power plants with respect to population distribution.

50-SG-D1 Safety functions and component classification for BWR, PWR and PTR.

50-SG-D2 Fire protection in nuclear power plants.

50-SG-D3 Protection system and related features in nuclear power plants.

50-SG-O3 Operational limits and conditions for nuclear power plants.

50-SG-O5 Radiological protection during operation of nuclear power plants.

50-SG-O4 Commissioning procedure for nuclear power plants.

50-SG-QA1 Preparation of the quality assurance programme for nuclear power plants.

50-SG-QA2 Quality assurance records system for nuclear power plants.

./..

50-SG-QA7 Quality assurance organization for nuclear power plants.

50-SG-QA 10 Quality assurance auditing for nuclear power plants.

3. REGULATORY ORGANIZATION

3.1. Organization of TAEK

It is regarded as essential that a government embarking on or implementing a nuclear power programme establish a regulatory body (2). It is recognized that in spite of its relatively small size, it must be able to deal with the somewhat more difficult aspects of supply, construction and operation in a developing country (3).

In Turkey TEAK is responsible for defining safety in respect of all nuclear activities and conditions governing the granting of authorizations to nuclear installations.

In January 1975, a Committee on Reactor Safeguards was set up to implement the licensing. This Committee is entrusted the task of approving supporting documents, reviewing, inspecting and, advising TAEK on the all stages of licensing.

The present organization of TAEK Contains a Nuclear Safety Department reporting to an Assistant Secretary General for Nuclear Safety. This department is entrusted the task of providing skilled main power to the Committee on Reactor Sefaguards. The Nuclear Safety Department includes the following Branches:

- Assessment and Licensing
- Regulatory Inspection
- Regulation and Guide Development
- Radiological Protection

./..

The establishment of a branch for Nuclear Materials Safety and Safeguards is recently proposed to TAEK. It appears little work for this branch at this stage. But it is felt that someone should be allocated the responsibilities for following progress and requirements for safeguards. Fig.2 shows the present organization for the regulatory body of Turkey.

It should be noted that during the initial period of the nuclear power programme, the regulatory body in Turkey was not an effective body. Presently there are 2 staff members in Assessment and Licensing Branch, 1 staff member in Regulations and Guides Development and 1 staff member in Radiological Protection.

At this stage, 8 members of Committee of Reactor Safeguards, including the chairman who is a commissioner, are providing an effective manpower for regulation and guide development. For the works of the Radiological Protection Branch, which is responsible for the safetly use of the radioisotopes and the other radiation instrumentation, the staff of the Ankara Nuclear Research Center are providing additional manpower. However this is not the case for the Assessment and Licensing Branch which bears the main task requiring manpower, since it is, in effect, an executive branch.

3.2. Expertise Required for the Regulation of Nuclear Power Plants

Considering the imminent decision to construct the first nuclear power station, it should be noted that, the present manpower of the Regulatory Body in Turkey is not satisfactory. Of course inexperienced young people can be trained abroad for this purpose, but the project is too far advanced to solve the immediate problem.

./..

It is, therefore, strongly recommended that the Turkish authorities act to quicken the creation of a more effective regulatory body and attract existing experienced people into regulatory body for this work.

In this situation the regulatory body should make maximum utilization of consultants and advisory groupes to compensate for the limited number of permanent regulatory staff. However, initially a staff of at least five or eight highly experienced persons is a minimum needed to staff a basic organization (1). But this staff should have broad technical abilities and engineering experience together with good judgement of safety.

Design assessment and the Contents of the Safety Analysis Reports Covers 19 areas with different expertise associated with it (4). However, this can be squeezed into 12, for the initial period, as shown in Fig.3. The numbers gives in brackets on this figure refer to the USA Standards Format and Contents of Safety Analysis Reports.

Therefore it is anticipated that the following staff members with associated background could be immediately appointed to the Nuclear Safety Department:

1. Nuclear Eng. with good background in Reactor Physics
1. Structural Eng. with nuclear knowledge
2. Mech. Eng. with M.S. degree in Nuclear Eng.
1. Eng. with B.S. in electronics and M.S. degree in Nuclear Eng.
1. Health Physicist with knowledge in Wast Management
1. Q/A Eng. with Broad experience

7 Total

This staff is sufficient only for the first nuclear plant. As the nuclear power programme grows the situation may arise wherein the regulatory staff could be required to perform review and assessment inspection, and regulation and guide development

activities simultaneously. If the nuclear programme is to include licensing and operating of 5-7 power reactors, as a minimum, 50 full time regulatory staff is necessary (2).

It should be noted that during the initial period of the regulatory body, the work load for the domestic groups such as Nuclear Engineering Department of Çekmece Nuclear Research Center (ÇNAEM) and Institute for Nuclear Energy of the Technical University of Istanbul could be very helpful.

4. CONCLUSIONS

It can be concluded that, at this stage, Turkey have not the proper balance and quality of skills required for the organization of Regulatory Body. The main difficulties are the lack of proper appreciation of the safety problems and the lack of experienced people available in Turkey.

The major regulatory functions suggest an initial organizational structure such as that shown in Fig.2. Its technical staff should be well-balanced, possessing, or having ready access to the expertise listed in Fig.3.

It is anticipated that, ÇNAEM, Institute for Nuclear Energy and other organizations will provide some of the expertise listed. Therefore, during the initial phase, of the project, the Assessment and Licensing Branch of regulatory body could consist as few as 7 individuals possessing broad technical expertise as shown in Fig.4.

In this situation the regulatory body should make maximum utilization of consultants and advisory groups to compensate for the limited number of permanent regulatory staff. Consideration must be given to the availabilty and proper use of technical

./..

assistances. Some advanced countries have established bilateral approaches to safety co-operation and assistance to developing countries. Turkey must be use effectively these advices and assistances as part of an overall co-operated programme to facilitate the necessary transfer of regulatory capability (5).

R E F E R E N C E S

(1) ROSEN M., IAEA Staff Member, "Nuclear Power Safety", Report to the Government of Turkey, WP/5/1457, 12 March 1979.

(2) CODE OF PRACTICE: "Covernmental Organization for the Regulation of Nuclear Power Plants", IAEA Safety Series 50-C-G (1978).

(3) ROSEN M., "The Critical Issure of Nuclear Power Plant Safety in Developing Countries", IAEA.

(4) WRIGHT H.A., IAEA Staff Member, "Reactor Safety Analysis", Report to the Government of Turkey, WP/5/1336, 4 May 1978.

(5) ROSEN M., IAEA Staff Member, "Nuclear Power in Developing Countries : The Transfer of Regulatory Capability", IAEA Bulletin Vol.21 No.2/3, June 1979.

(6) AYBERS M.N., Technical University of Istanbul, "Problems Associated with Nuclear Energy Utilization in Developing Countries", Institute for Nuclear Energy Bulletin No.23, 1975.

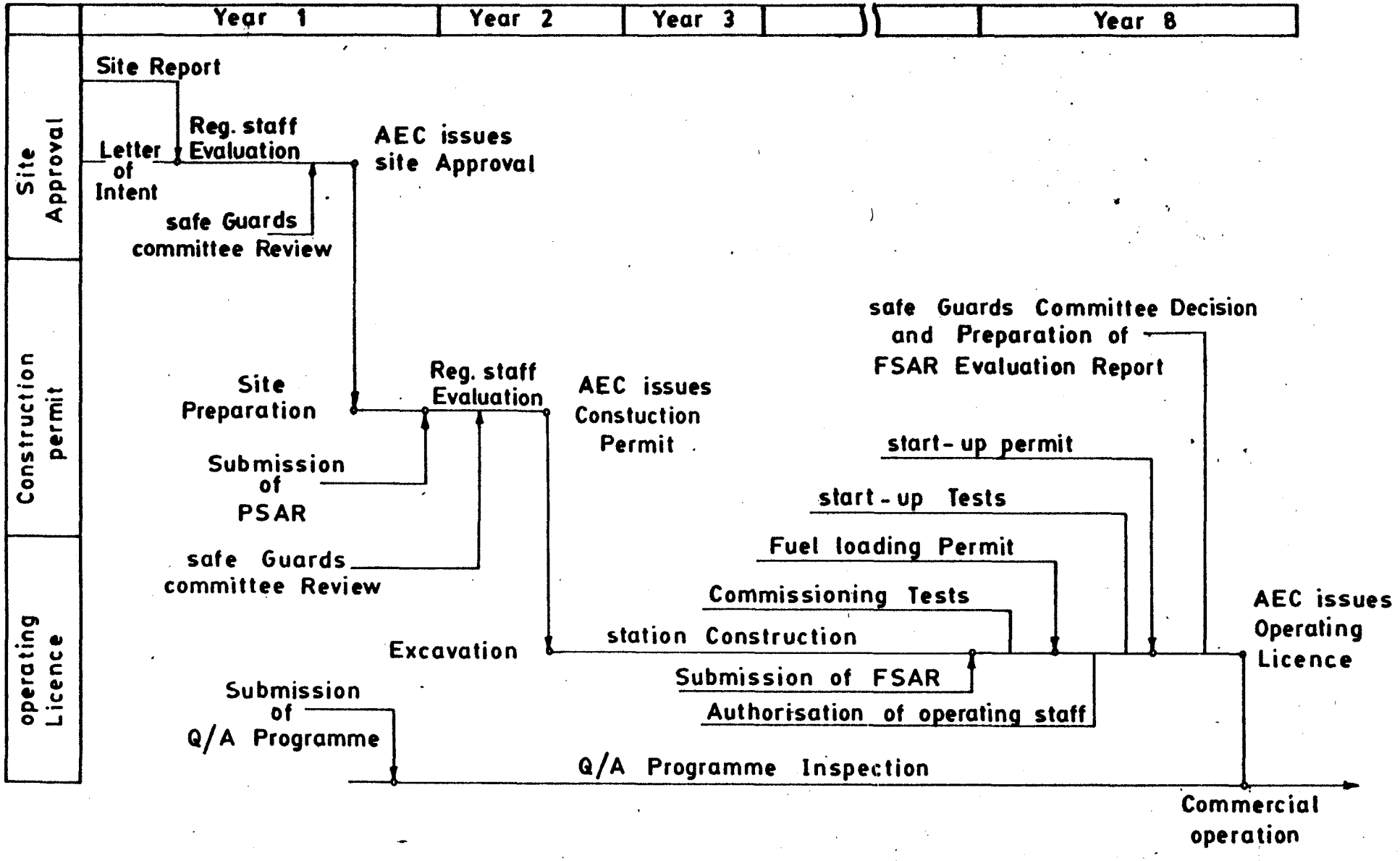


Fig. 1 Licensing Process in Turkey

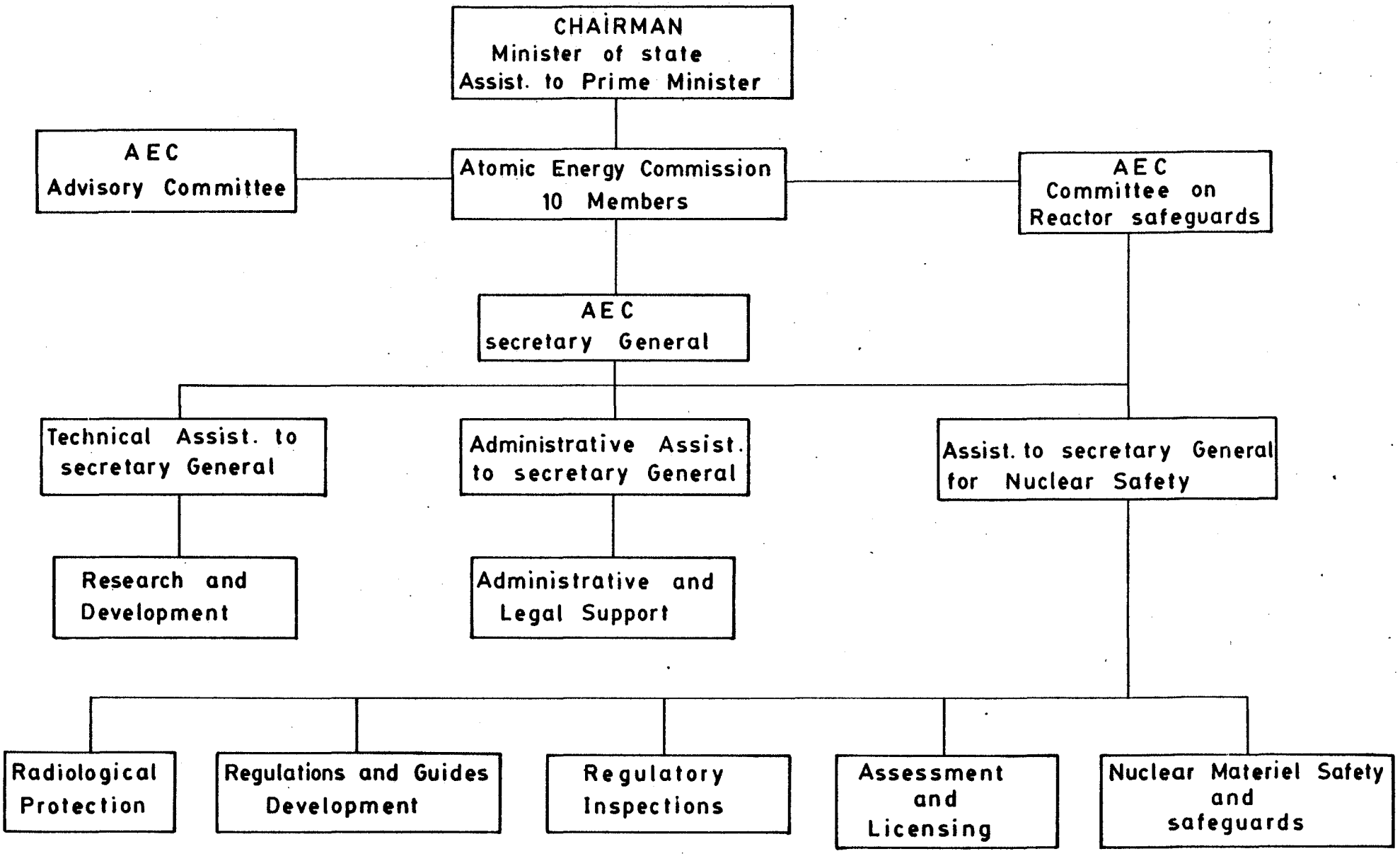


Fig. 2 ORGANIZATION FOR THE REGULATORY BODY IN TURKEY

Fig. 3 · Expertise Required for the Assessment and Licensing of the first Nuclear Power Plant

Assessment Areas	Appropriate Specialists
1. Structures (3.0)	Civil Engineering
2. Reactor (4.0)	Nuclear Eng./Reactor Physicist
3. Reactor coolant System (5.0 except 5.6)	Mechanical Eng.
4. Containment Systems (6.0)	Nuclear Eng./ Mech. Eng.
5. Auxiliary Systems (9.0)	Nuclear Eng./ Mech. Eng.
6. Instrumentation and Control (7.0 and 5.6)	Control and Instr. Eng.
7. Electrical Power (8.0)	Elect. Eng.
8. Steam and Power conversion System (10.0)	Mech. Eng.
9. Radiation Protection and Waste Management (11.0 and 12.0)	Health Physicist
10. Initial Test and Operations (13.0 and 14.0)	Nuclear Eng. (Broad Background)
11. Accident Analyses (15.0)	Nuclear Eng./Reactor Phys.
12. Ouality Assurance (16.0)	Q/A Expert (Broad Experience)

Fig. 4

Expertise Required in the initial phase of the Nuclear Power Project to staff the Assessment and Licensing Branch

Number	staff member	Required Expertice
1	Nuclear Eng. or. Reactor physicist	Good Knowledge in Computer codes
1	Civil Eng.	Good Knowledge in structural Eng. and Seismicity
2	Mechanical Eng.	With M.S. Degree in Nuclear Eng.
1	Electrical or Electronics Eng.	With M.S. Degree in Nuclear Eng.
1	Health Physicist	Waste Management
1	Q/A Engineer	Broad Experience in industry
7	Total staff Members	

Second Period of Discussions on Session II

N. Aybers (Turkey) (To H. Matulla)
Did you perform reactor internal vibration tests?

H. Matulla (Austria)
During the 'non nuclear hot test' there were some tests in connection with the internal pumps.
There were vibration tests planned for the nuclear startup of the plant. But the referendum stopped this and they were not performed.

J. van Daatselaar (Netherlands) (To H. Matulla)
With respect to the second failure tables: In the analysis carried out to show that there is no unallowed influence on a required safety function, do you apply a probabilistic approach?

H. Matulla (Austria)
No, this analysis is done in a deterministic way. It is possible that there can be some so-called 'second failures' if they have no unallowed influence on the required safety capacity.

J. van Daatselaar (Netherlands) (To J. Laaksonen)
In your paper you say a conservative design with safety margins or multiple redundancies was often preferred. Were you able to require design changes and how did you improve the safety margins?

J. Laaksonen (Finland)
Yes, especially in the Loviisa plant we required plenty of design changes as compared to the original reference plant built in the Soviet Union. All ECC systems, reactor containment and most of the instrumentation are totally of new design. The safety is tried to be improved by using four redundant trains in most safety systems. As another example I can mention the reactor core where the linear heat power is much less than in western plants and also clearly less than in Soviet plants. The latter is due to even power distribution in Loviisa and improved in-core instrumentation to verify that distribution.
The TVO plants are of third generation of ASEA-ATOM plants and there we have not required any drastic changes but only some modifications in details.

R. Gausden (UK) (To J. Laaksonen)
Could you give some idea of the criteria you use to determine whether backfitting of plant is required.

J. Laaksonen (Finland)
We have no established criteria for backfitting. If some weak points are revealed in the operating plants it is always a matter of engineering judgement to decide if corrective measures are needed. As a common rule we have required backfitting to be done in all cases where it clearly improves the safety and can be done in a reasonable way.

C. Pérez del Moral (Spain) (To J. Laaksonen)
In your paper you said that few members of your Organization's staff have quit it to take other jobs. I'd like to know what in your opinion are the most important factors behind this achievement.

J. Laaksonen (Finland)
I am not sure what is the most important reason, but I can suggest some factors that may have contributed. First is the good spirit inside the regulatory body, most people find their work interesting and the working methods give them enough motivation to do their very best. Another reason might be a general depression period since 1975, after that year there have not been many vacant places for engineers in our country. Also in the nuclear field no big projects have been started since 1975. Some influence has also been exerted by the fact that there is no big difference between salaries in the regulatory body and in the industry.

M. Perelló (Spain) (To J.M. Oury)
Après l'exposé de M. Oury et vu l'intéret pour l'évolution des centrales nucléaires de puissance, du point de vue de la sûreté nucléaire, il semble qu'il est préferable pour les pays de disposer d'un seul type de centrale. Mme. Pérez del Moral a répondu qu'en Espagne une telle situation était impossible car c'est l'industrie privée qui décide du type de centrale à construire. Etant donné que la France a décidé de ne construire qu'un seul type de centrale PWR, la sûreté de ces évaluations en bénéficiera. Ma question à M. Oury est la suivante: les services de sûreté nucléaire français ont ils été consultés et ont-ils donné leur accord sur le plan de la sûreté nucléaire aux autorités françaises avant que la décision ne soit prise de n'utiliser que des réacteurs PWR et d'abandonner les réacteurs gaz-graphite. Si oui, est-ce que M. Oury, en tant que membre du SCSIN, considère

que les réacteurs PRW sont plus sûrs que les gaz-graphites? ou bien simplement que leur degré de sûreté est suffisant?

J.M. Oury (France)
(1) Il va de soi que la décision d'engager la réalisation d'une centrale nucléaire quelle qu'elle soit doit faire l'objet d'un accord des autorités responsables de la sûreté; cet accord est d'ailleurs l'objet des procédures rigoureuses prévues en France et décrites dans l'exposé.
(2) En matière de PWR, cet accord a existé dans le passé comme en témoignent les autorisations de création et de fonctionnement déjà délivrées. A noter toutefois qu'existent en France d'autres types de réacteurs (GG ou rapides notamment) et que l'autorisation de création de l'un d'entre eux (Creys Malville) est postérieure à l'accélération du programme éléctro-nucleaire en matière de PWR.
(3) Il n'y a donc pas decision de la part des autorités de sûreté d'imposer l'unicité du type de réacteurs: les exploitants sont les premiers responsables de leurs installations et ont donc le choix de leurs propositions. Il y a par contre volonté des autorités de sûreté de tirer le meilleur parti du point de vue de la sûreté de la proposition de standardisation qui leur est faite et volonté également d'en limiter les inconvénients (essentiellement préserver l'ouverture au progrès technique).
(4) Quant au niveau de sûreté entre filières, je crois toute comparaison extrêment difficile sinon impossible. L'essentiel est que chaque filière atteigne un niveau de sûreté au moins satisfaisant et en fait aussi élevé que raisonnablement possible.

F.J. Turvey (Ireland) (To J. Laaksonen)
I notice that only one person is employed on Rules and Standards within the Department of Reactor Safety. Although you have used standards of other countries I am surprised that you only employed one person. Could you comment, please?

J. Laaksonen (Finland)
That one person only co-ordinates the rule and standard work and keeps an eye on new standards published in other countries. The actual work to develop own rules and guides is done inside the technical groups.

P. Giuliani (Italy) (To J. Laaksonen)
Which authority does perform site reviews in Finland?

J. Laaksonen (Finland)
The site review is performed with the help of another IRP department called the Research Department. They carry out

environmental research on plant sites. Inside the Department of Reactor Safety the site aspects are the responsibility of the radiation protection group.
Generally I can say that the site problems are not as difficult as in most other countries. We have no earthquakes and no hurricanes, the plants can be located in remote sites far from large towns, there is no dangerous industry or air traffic close to the sites and the availability of cooling water does not cause any problems.

J.S. MacLeod (UK) (to J.M. Oury)
Is there a danger with too much standardisation that a too large a portion of a power generation system may be involved?

J.M. Oury (France)
Il s'agit principalement d'un risque économique. Sur le plan de la sûreté, il est certain que cette question est liée à l'une des difficultés exposées dans la communication: si un problème se trouve posé sur une tranche, il est posé d'emblée sur toutes les tranches.
Mais cette difficulté doit être nuancée par plusieurs remarques:
(1) Comme TMI l'a montré, quand il survient quelques chose de vraiment important tous les réacteurs sont concernés, et - sauf exception - des enseignements sont à tirer pour tous: multiplier les différences de détail n'apporterait à cet egard aucune facilité.
(2)La standardisation facilite la prise en compte de l'expérience: un défaut de conception ou de réalisation 'précurseur' éventuel d'un évènement plus grave peut être détecté à la 20ème tranche, et la modification correspondante bénéficie à toutes les tranches précédentes.

SESSION III

TECHNICAL BASES FOR THE REGULATORY REVIEW

Chairman: G. Tenaglia

Scientific Secretary: P. Lebouleux

TECHNICAL BASES FOR REGULATORY REVIEW ;

COMPARISON OF PRACTICES, STANDARDS AND GUIDES

W. Vinck ; H. Maurer ; G. Van Reijen
Commission European Communities
Brussels, Belgium

ABSTRACT

First the scene is set with regard to the national licensing and regulatory scenes and with regard to correlations between

- regulations and industrial standards
- national and international developments.

The purpose of harmonisation efforts and ongoing activities, especially within the EC, are highlighted.

For the purpose of being sufficiently specific, three specific areas are dealt with as examples, explaining how actual application of practices and criteria in EC Member States correlate (or not) to one another including the points of convergence and divergences : protection against aircraft crash, seismic effects, protection against fuel handling accidents.

Conclusive remarks deal with the origin in and relative importance of discrepancies in safety practices and criteria.

1. Scene-Setting

1.1 The national scenes in the nuclear area : licensing, regulation, standard-setting and their connexions

1) General background

The licensing procedures in the various countries have developed - and are continuing to develop - along different lines, depending on

- the political and administrative structures and the laws applicable in each country;

- the organizational characteristics of each country and its regions.

These procedures are the vehicle for the development and/or application (again with variations) of the technical practices and methods concerning nuclear safety as well as the relevant safety requirements mostly spelled out in regulations, criteria, guides, standards and codes for site, plant structures, systems and components which we will call in common "rules".

As long as there is no sufficient politically structured unity in Europe it seems a vain exercise to optimize community efforts towards harmonizing legislation and administrative procedures.

Therefore emphasis in community efforts has been placed upon the technical, aspects, although the procedural aspects often correlate with the technical aspects.

Therefore let us now deal with technical safety practices and "rules" aspects, more specifically the regulatory aspect.

2) Technical Methods and "Rules"

Against a background of evolution over the last twenty years in a rather wide range of safety analysis methods (e.g. M.C.A. concepts, successive barriers concepts, defense-in-depth concepts, deterministic approaches, quantified probabilistic approaches), more specifically since 1965 and initially in certain of the most advanced countries in the nuclear field, systematic efforts to develop "rules" have been made.

How can the rôle to be played by these rules be defined ? The rules represent a systematic and disciplined codification of the engineering and, simultaneously, a review of the lessons provided by experience.

If these rules are applied by the industrial architect who designs and by those who construct and operate the nuclear power station, they can contribute to the confidence that the operator can have in the reliability, the availability and the safety of the plant. In addition, they contribute to decreasing the cost of design and of manufacture and to facilitating the "planning" of manufacturing and construction.

This codified good practice (rules) provides the licensing authorities and the safety and control organizations with a high degree of confidence that the design basis and the performance requirements on the equipment on which they based their safety analysis and their approval will effectively be achieved.

Rules can be evolved by a regulatory authority and/or associated safety and control organizations, in which case they find expression in mandatory (obligatory) requirements or indicative (quasi-mandatory) requirements. As example one may quote 10 CFR 50, Atomgesetze, Ministerial decrees as mandatory and the USNRC regulatory guides or KTA-Regeln developed by joint effort of authorities, utilities, and vendors however) as indicative rules.

On the other hand, they may evolve on a voluntary basis, i.e. by a joint effort on the part of experts of the electricity producers, industrial architects and builders, and the licensing and regulatory authorities and/or the associated safety and control bodies (example : ANS,ANSI, ASTM, ASME-codes and standards, D.I.N.). They are then usually called industrial "standards".

Such rules have to be adapted to take account of advances in technological knowledge (e.g., the results of safety research) and the evolution of safety methods (e.g., the use of probabilistic methods of analysis).
This means that they have to be periodically revised.

1.2 The international scene in the nuclear area : development of rules (regulatory and industrial standards); harmonization of practices and rules.

General

If the need to develop nuclear power is recognized and the industries associated with it, it seems essential to evolve harmonized cross-frontier approaches and techniques in nuclear health and safety matters (covering both the technological and radiation protection aspects).

Indeed :

- there is no reason to allow the development of different overall levels of safety to which individuals of different nationalities are exposed ;

- in view of the growing international exchange of projects and technical equipment (structures, systems, subassemblies and individual components), disparities in the rules, i.e. in the regulatory requirements and in industrial standards, constitute a barrier to industrial development and energy production;

- the development of standardized methods to deal with safety problems and their environmental implications for the design, fabrication and assembly of equipment is bound to shorten lead times for the setting-up of nuclear power stations, and speed up their putting into service.

The nature of the rules and the way in which they have been developed at national level show that, whatever merits the systematic approaches applied may have, the outcome is a very complex situation which is rendered all the more difficult in view of harmonization by the magnitude of the problem and by the national efforts dealing with it. Any international attempt at harmonization which is not sufficiently selective and judicious in the choice of options and priorities would - in my view anyhow - fail to make any impact.

The three most noteworthy approaches (but each with a different scope and purpose) - i.e. at least in the technological area (the radiation protection area is widely covered by the I.C.R.P. and also within the EC*) - to the problem

*Basic Standards

at international level are those undertaken by the International Atomic Energy Agency (IAEA), the International Standards Organisation (ISO) (together with its sister organization the International Electrotechnical Committee (IEC)), and the European Communities (EC).

In particular, the harmonization efforts by the Commission of the European Communities will be highlighted here.

1.3 The Commission of European Communities

A systematic effort to achieve progressive harmonization of safety practices and requirements for nuclear power plants, more specifically for light water reactors, was launched in 1973. A permanent working group on "Safety methodologies, criteria, standards, etc." (WG N° 1) was set up for this purpose, in which the licensing (and regulatory) authorities and associated safety and control organizations are represented on one hand and the utilities and vendors on the other.

The basic support, at the political level, for these harmonization efforts was provided by the European Community Council of Ministers Resolution on nuclear safety technology of 22 July 1975 (1) which amongst others and in short :

- calls for an effective cooperation and for a strengthening of effort at harmonization of safety requirements and criteria within the European Community;
- requests the member states to notify the Commission of any draft laws, regulations or provisions of similar scope concerning the safety of nuclear installations in order to enable the appropriate consultations to be held at Community level at the initiative of the Commission;
- provides that, at the appropriate time, Community recommendations shall be issued (Article 124 of the EURATOM Treaty*) on those subjects which lend themselves best to a harmonized approach.

The main tasks of Working Group N° 1 can roughly be summarized as follows :

- Dialogue, discussion, identification and exchange of information and documentation on safety methodology, criteria, codes and standards and specific LWR safety problems applicable and/or under development in the various member countries.
- Identification of divergencies, similarities, common requirements, reasons for divergencies and establishment of synthesis reports in view of the elaboration of recommendations in areas and on items of common interest and considered mature for that purpose.

It is also worthwhile to note that working group members participate in the consultation on drafts of internationally established codes and guides, especially those within the IAEA-NUSS program.

The working group has set itself an order of priority for dealing with the various subjects and has in the course of 1977 worked out a revised working

---

*) Title III, Chapter I, Section III, Article 124 of the Treaty establishing the European Atomic Energy Community.

scheme for its 2nd phase of work, which was started in the beginning of 1978.

The terms of reference of Working Group N° 1 were adapted to the revised work programme by the insertion of the provision that the next stage should start with the selective formulation, according to necessity and in an agreed order of priority, of recommendations within the meaning of Article 124 of the Euratom Treaty.

The subjects which the working group handled to the present time can be classified in the following main sections :

- fundamental concepts
- hypothesis and methods of analysis concerning the safety of the NPP as a whole
- safety specifications for systems and components.

Before any detailed recommendations (Article 124 of the EUR Treaty) are prepared, a sequence of the range of recommendations will be prepared. First priority was given to the general safety principles recommendation which is actually underway.

It is now considered by the authorities that recommendations should be concerned with safety philosophy rather than design detail.
Arising from this, the development of recommendations lies mainly in the hands of the safety authorities within Working Group N° 1 in consultation with the utilities and vendors.

After the elaboration of a general safety principles recommendation, specific topics, for which progress in comparative scrutiny has been sufficient in the area of protection against external and internal events may be amenable to recommendations.

The Three-Mile Island events will undoubtedly influence some of the approaches taken in the Working Group N° 1 activities.

The overall subjects which were covered by the Working Group until the present time are presented in table I (Appendix).

Out of the numerous subjects dealt with as outlined in this table, we have picked on three in which noticeable progress was made in understanding the various position taken and in delineating as clearly as possible the convergences and divergencies in approaches, namely for :

- protection against one man-made external effect, i.e. aircraft crash
- protection against a natural external effect, i.e. seismic effects
- protection against an internally originated accident conditions, i.e. fuel handling accidents.

2. Protection of Nuclear Power Plants against Aircraft Crash

2.1 General safety criteria of EC-member countries mention the need to protect nuclear power plants against aircraft crash. Specific guides to this respect are under development and discussed amongst the experts of the EC-countries in view of harmonization of presently applied practices.

In particular the following considerations are important in this context :

- to minimize the risk of aircraft crash on nuclear power plants
- to define exclusion areas around airports and
- to prohibit flying in the vicinity of nuclear plants.

2.2 Probabilistic considerations

In some countries the probability of crashing for each type of aircraft is evaluated separately for each site and in dependence of this probability a decision is taken on design of the plant against such a crash. Generally these countries define that for each unit the probability of inacceptable release is less than a certain limit. These countries calculate the mentioned probability by multiplying the crash probability with the conditional probability of structural damage. So for each site the crash probability is calculated and by protective measures the conditional probability of structural damage is reduced so far that the product of both probability values is below the limit. In other countries all nuclear power plants have to be designed against the effects of crash of certain types of airplanes, e.g. by protection of structures against penetration and/or by physical separation of diverse safety equipment. In most countries no general backfitting requirements exist.

2.3 Siting (with regard to aircraft crash protection)

There are no important differences in siting policy between E.C.member countries. In all countries exclusion zones of 8-10 Km from airports are prescribed or recommended. This exclusion zone exists also in countries, where all nuclear power plants have to be designed against aircraft crash in view of the risk of failure of the protection.

2.4 Arrangements with airforce and civil aviation

In most E.C.member countries there is general agreement with airforce and civil aviation not to fly close to nuclear power plants. In particular it has to be avoided that nuclear power plants are used as an approach target. Attention is now paid to the question of beaconing of nuclear power plants.

2.5 Consequences for plant design

In those European countries in which as a consequence of probabilistic considerations it was decided to protect every nuclear power plant against aircraft crash certain types of airplanes are used as reference planes for defining the load-time functions for designing the plant structures.

In Germany and the Netherlands nuclear power plants are designed to resist the impact of an aircraft with a weight of 2.10^5 N crashing at a speed of 215 m/s (Phantom RF 4 E has the characteristics mentioned).

In the UK a weight of $2,5 \cdot 10^5$ N is considered (MRCA). This load function will cover a wide range of military and civil aircrafts. The impact area is assumed to be 7 m^2 (compared to civil aircrafts with impact velocities of 100 m/s and impact areas of 18 - 37 m^2 - depending on the surface if flat or spherical).

Parts of the aircraft, as e.g. engines, wings etc. can become secondary missiles as a result of the crash. (The load resulting from a military aircraft engine being $1,8 \cdot 10^4$ N at an impact velocity of 100 m/s and an impact area of about 1,5 m^2). In the design against aircraft crash these loads are combined with secondary loads, plant malfunction, debris, et

As far as the computation methods applied are concerned they are in general more or less based on penetration experiments of projectiles in concrete, most of which were performed to investigate the penetration of bullets and bombs into reinforced concrete target slabs. Thus the results may not be directly applicable to the design of nuclear power plants against aircraft crash. This was the reason that in France other correlations and formulas were developed.

A comparative study of calculation methods for the resistance of structures against penetration and for the overall stability of structures is recommended in view of a harmonization of the practice in this field.

3. Protection of Nuclear Power Plants against Seismic Effects

The effects of seismicity, as they are observed in European countries may, under certain circumstances, lead to significant nuclear hazards, so that every effort must be undertaken to protect nuclear power plants against such abnormal natural hazards. This is necessary not only in countries with high seismicity, but also in regions with medium or even low seismicity.

Consequently, it is necessary to identify those structures, systems and components of a nuclear power plant which must be designed to maintain their safety related function under the maximum potential seismic load, and to define the methodology which must be followed to ensure an acceptable earthquake resistant design.

For this reason it was considered worthwhile to discuss the problems of protection of nuclear power plants against seismic effects within the European Working Group on "Safety of Water Cooled Reactors - Methodology, Codes and Standards" (Working Group N° 1).

Most Member States of the European Communities base their anti-seismic safety and design concept and even specific design parameters on the current concept of the USA which has been developed with respect to the specific characteristics of the seismicity of North America. There are, however, important discrepancies between the two continents in the seismological and geological conditions as well as in the general design demands.

This is why many European countries realise now that they have to develop their own specific safety concepts adapted to the typical seismological and geological conditions in the respective countries.

The work performed within the activities of Working Group N° 1 is based on an inventory of the applied national practices, the existing specifications, regulations, and guidelines applied in the design, construction, and safety assessment of structures, systems and components to withstand potential earthquake effects. These specifications and guidelines are compared with those for the USA and Japan and due notice is taken of the work performed within other international organisations as are ISO and IAEA. Points of agreement were identified and divergencies discussed in particular with reference to the US practice.

As a basis for design two different reference earthquakes are defined; the first more severe earthquake (the safe shutdown earthquake SSE *) is associated with the safe shutdown of nuclear power plant, while the second (the operating basis earthquake OBE *) is associated with its reliable operation. The procedures applicable and actually in use for establishing the reference earthquakes for a given site are the deterministic approach as well as the probabilistic approach. Both approaches rely on subjective judgements based on seismological and geological experience.

The SSE is based upon an evaluation of the maximum earthquake potential considering the regional and local subsurface material. It is the earthquake which produces the maximum vibratory ground motion for which those structures, systems and components are designed to remain functional and to assure :

(a) The integrity of the reactor coolant pressure boundary,

(b) the capability of shutting down the reactor and maintaining it in a safe shut-down condition, or

(c) the capability of preventing or mitigating the consequences of accidents which could result in potential offsite exposures comparable to the NRC guideline exposures of 10 CFR 100.

Since the SSE is related to the maximum earthquake potential at the site, it has a low probability of occurrence during the life time of a nuclear power plant. Nevertheless, in most European countries the OBE is determined in dependence of the SSE assuming that the maximum vibratory ground acceleration of the OBE shall be at least one-half of the maximum vibratory ground acceleration of the SSE.

Only according to French practice the larger reference earthquake which is called "Séisme Majoré de Sécurité SMS" is determined by adding a certain safety margin to the intensity of the smaller reference earthquake which is named "Séisme Maximal Historiquement Vraisemblable SMHV" (Maximum Plausible Historical Earthquake). The safety margin applied is usually one degree of the macroseismic intensity measured with the MSK-scale.

---

* according to US definitions

At present, preference is given to the deterministic approach in the evaluation of both, the SSE and OBE, although the assessment of the OBE with statistical and probabilistic methods on the basis of available data in Europe appears feasible.

At a recent discussion under experts collaborating with working group N° 1 it came out that for the definition of the lower level earthquake no agreement at all seems to be possible on a deterministic basis, there may, however, be a chance to agree on a probabilistic basis.

4. Protection of Nuclear Power Plants against fuel handling accidents

A comparison of the different methods and procedures applied by European countries, in the analysis of the fuel handling accident was performed within the activities of Working Group N° 1 in order to investigate convergencies and divergencies in safety assessments. It can be demonstrated that existing differences in the assumptions of the basic conditions (as are decay time, or mode of operation) are only of minor importance. The assessment of a fuel drop is in all European countries based on the damage of all fuel rods (of one fuel element). However, here appears to be a tendency at least on some European countries to assume that only the fuel rods of an outer array be damaged. This was also assumed for the evaluation of the radiological consequences of a German nuclear power plant. Also the basic assumptions concerning the quantity of fission products released from the fission gas plenum to the pool water differ from country to country, with the most conservative assumption in the US Regulatory Guide 1.25*. More realistic assumptions used in France or Germany are based on experimental results of iodine release (organic and inorganic) less than 10 % down to 1 %.

Minor differences are observed also with regard to the decontamination factors for storage pool water.
(Decontamination factor = $\frac{\text{activity release to pool water}}{\text{activity release to building atmosphere}}$)
there again the assumptions published in the US Regulatory Guide 1.25 being extremely conservative.

Off-site radiological consequences of the accident are limited by the construction of the storage building and the filter equipment of this building. In the USA these filters obviously are not mandatory. They are, however, provided in all European countries, also in those which apply the US rules and guides in general.

The design of the fuel handling devices have in all nuclear power plants the same objective : to avoid the fuel handling accident. The fuel handling transfer machines are designed to preclude mishandling, the designed safety features being as follows :

- Lifting of the fuel assembly is not possible until the grips are additionally locked mechanically.
  This locking mechanism must be redundant and diversified.

---

*(according to RG 1.25 the assumed releases are as follows :

noble gases	10 %
Kr-85	30 %
Iodine	10 %

- Lifting and travelling motions must be preclusive, the relevant locking mechanism being either positive or redundant.

- Travelling motion must be impossible until a prescribed minimum lifting height has been achieved.

An important difference exists in some European countries (especially Germany, France and Belgium) where the fuel handling machine must be protected against external events (e.g. earthquakes, airplane crash, explosion pressure waves).

The effects of such external events are analysed locally and the relevant structural components are so designed as to preclude fuel handling accident due to external events.

Other important divergencies exist with respect to the hypothetical fuel cask drop inside the storage building. These divergencies result from the different location of the fuel storage pool either within the containment as in German plants, or outside the containment in a separate fuel storage building as in US, French or Belgian plants. In older German plants the fuel cask is lowered directly into the spent fuel pool. In US, French and Belgian plants there are three different pools, the storage pool, the decontamination pool, and the loading pool (without fuel elements).
All this applies primarily to existing power stations, whereas the differences appear to be eliminated in German power stations under construction or planned where a separate fuel cask pool is foreseen within the containment next to the fuel storage pool.

5. Conclusive Remarks

Very often discrepant evolution can be noted, especially in the technological area (but there are also examples in the radiation protection area : e.g. present tendences with regard to application of publication 26 of ICRP and with regard to limits for professionnally exposed). They result from :

a) different vendor's engineering knowhow and inventiveness, as influenced by national licensing and regulatory requirements and by their clients (the utilities) interests;

b) from requirements imposed by the licensing and regulatory bodies that refer mainly to the relationship between the plant and its environment, i.e. the site-dependant characteristics (including external impact hazards : seismic effects, gas-cloud explosions, aircraft crash, flooding, acts of sabotage) and the related emergency planning provisions.

It is clear that, if the regulatory requirements are significantly different from one another, this on one hand is difficult to explain to the public exposed to a certain risk and on the other hand puts the respective vendors in front of varying situations in an international market.

Admittedly it is not easy to define when a requirement becomes significantly different (from the standpoint of health and safety and economically speaking) but efforts in assessing this on a quantitative basis (including the notions of risk and cost-effectiveness) merit more attention.

Also in many instances different levels of protection can be explained, or perhaps even justified on the basis of different site-conditions (e.g. seismicity, aircraft traffic density, population density, etc.) and on the basis of overall socio-economic conditions, but there are instances where this is not evident.

It is submitted for consideration that an important factor to be considered in the future is the respective situation of nuclear versus conventional major hazards industries.

However both these two latter connected considerations go beyond the scope of the present paper.

x

x x

The opinion expressed in the present paper do not necessarily reflect the views of the Commission of the European Communities.

Main reference material

(1) Council Resolution of 22 July 1975 relative to technological problems of nuclear safety.
Official Journal of the European Communities, N° C 185/1, 14 August 1975.

(2) Recommendations of the International Commission on Radiological Protection (adopted 17 January 1977)
ICRP Publication 26.

(3) M. BANAL
Harmonisation des règles de sûreté des centrales à eau légère.
UNIPEDE Congress at Warsaw, June 1979.

(4) W. VINCK
Regulatory practices and standards : the international scene and trends.
Paper presented at the International Conference on Regulating Nuclear Energy.
Brussels , 16-19 May 1978.

(5) W. VINCK et AL
Safety rules and practices, developments and incentives to international cooperation.
Salzburg Conference, May 1977. AIEA - CN 36/556 (IV)

(6) Practices and rules for nuclear power stations; the role of the Risk concept in assessing acceptability; W.VINCK , paper presented at CEC/CERD Seminar on Technological Risk;
Berlin, 1-3 April 1979.

(7) Harmonisation of nuclear Regulation in Europe ; W.VINCK, W. ESSLER;
Nuclear Engineering Int. October 1979.

(8) Unpublished reports developed in frame of CEC WG N°1 (Water Reactor Safety methodology, criteria, standards).

TABLE I

SURVEY OF WORKING GROUP N° 1 ACTIVITIES.

Generals

General safety principles and a sequence of recommendations

Quantitative methods applied in preparation and assessment of Safety Reports (frequency of event or risk concept)

Contents of safety reports
(Comparison of practices with regard to fault and accident conditions that determine the design of NPPs)

External hazards

Protection of NPPs against aircraft crash

Protection of NPPs against industrial environment hazards (external explosion including flammable vapour clouds)

Protection of NPPs against seismic effects

Protection of NPPs against floods.

Internal hazards

LOCA-ECCS

- Mechanical and thermal hydraulic effects

- Reference accident assumptions and radiological consequences

Spectrum of steamline breaks inside and outside containment

Turbine missiles

Coolant pump flywheel integrity

Fuel handling accident

Anticipated transients without scram (ATWS)

Safety, Design and Operational Provisions

General design criteria

Quality assurance during operation (comparison of Q.A. requirements)

Reactor coolant pressure boundary
(Correlation study on criteria, codes etc. for primary boundary)

Overpressure protection of primary circuit

Inservice inspection of primary pressure boundary

Reactor protection systems and control (correlation study on practices and criteria)

Containment structure and engineering safeguards
(Methods and procedures applied in containment leak testing)

Protection of NPPs against loss of electric power supply
(emergency power supply)

Fire protection

Siting and Emergency Planning

- Site suitability criteria, densely populated areas etc.

- Emergency planning technical input data (practices, trends)

Incidents reporting Requirements and Systems

DEVELOPPEMENT DE LA REGLEMENTATION TECHNIQUE DE SÛRETE NUCLEAIRE EN FRANCE ET SON UTILISATION DANS LA PROCEDURE D'AUTORISATION

P. Lebouleux

Commissariat à l'Energie Atomique
Fontenay-aux-Roses

Le Commissariat à l'Energie Atomique a regroupé initialement toutes les activités dans le domaine nucléaire y compris celles relatives à la protection et à la sûreté nucléaire. Avec l'apparition d'autres partenaires, les pouvoirs publics ont mis en place une réglementation nationale relative aux installations nucléaires à partir de 1963. Ces règlements prévoient notamment l'introduction de prescriptions auxquelles doit se conformer l'exploitant dans les autorisations de création des installations et la mise en place d'une réglementation technique générale (RTG) en matière de sûreté nucléaire. La RTG comprend trois niveaux hiérarchisés suivant leur degré de généralité. Dans la procédure d'autorisation on s'appuie sur les textes réglementaires publiés, mais également en tant que de besoin sur les décisions ou recommandations des autorités de sûreté et sur les projets de règlement suffisamment élaborés.

Initially, the Commissariat à l'Energie Atomique was the overall structure into which all nuclear activities were incorporated, including those connected with protection and nuclear safety. As other partners appeared, the authorities since 1963, have laid down national regulations relative to nuclear installations. These regulations, more particularly, provide for the addition of prescriptions, with which the applicant must comply , to licenses concerning the setting up of installations, and for the establishment of general technical regulations, or "réglementation technique générale" (RTG), in the matter of nuclear safety. The RTG, which is being drawn up, comprises three levels, according to the degree of generality involved. In the licensing procedure, one usually relies on published regulatory documents, but also, if need be, on safety authorities guidelines or recommandations and on sufficiently matured regulatory projects .

I - Cadre du développement de la réglementation technique générale en matière de sûreté nucléaire

1. Le Commissariat à l'Energie Atomique créé en 1945 a regroupé d'abord toutes les activités nationales dans le domaine de l'Energie Atomique. Parmi les missions qui étaient confiées à cet établissement, on relevait à côté de celles de développer les recherches et d'assurer le développement en vue de l'utilisation de l'énergie nucléaire, celle de "proposer les mesures propres à assurer la protection des personnes et des biens contre les effets de l'énergie atomique, et de contribuer à leur mise en oeuvre". On voit ainsi que dès le début du développement de la recherche et de l'industrie nucléaire française, les problèmes relatifs à la protection et à la sûreté nucléaire ont été au premier plan des préoccupations des pouvoirs publics.

A partir de 1957, Electricité de France et l'industrie française font leur entrée dans le domaine nucléaire pour développer d'abord la filière à Uranium Naturel - Graphite Gaz, puis à partir de 1970 le programme de grande envergure des centrales à eau sous pression construites sous licence américaine. C'est également vers cette époque que les différents partenaires français, dans le cadre d'une coopération internationale, s'engageaient dans la filière des surrégénérateurs.

L'accroissement de l'effort nucléaire français allait de paire avec la mise en place de dispositions réglementaires visant à assurer la sécurité des installations nucléaires.

D'abord au CEA était créé en 1960 une "Commission de Sûreté des installations atomiques" ainsi que des sous-commissions spécialisées, dont l'une dans le domaine des réacteurs nucléaires. La Commission est notamment chargée "d'officialiser , en liaison, le cas échéant, avec les organismes nationaux et internationaux qualifiés, les normes de sûreté des installations (...), de délivrer les certificats de sûreté pour la construction des piles et des installations (...), de délivrer les licences d'exploitation des piles et des installations au moment de leur mise en service (...)".

En annexe des notes de création de la Commission et de ses sous-commissions, on trouve le descriptif sommaire du rapport de sûreté d'une pile atomique.

Cette Commission est un organe interne au CEA et de ce fait, n'a aucun droit de regard sur les installations d'EdF, cependant, à titre officieux, les rapports de sûreté des premières centrales graphite-gaz ont été remis à la sous-commission de sûreté des piles du CEA.

2. A partir de 1963, les pouvoirs publics mettent en place la réglementation nationale relative aux installations nucléaires. Dans le domaine de la présente communication, la réglementation prévoit que :

- les installations nucléaires de base (INB), c'est-à-dire les réacteurs et les autres installations nucléaires importantes, sont autorisées par un décret pris par le Premier Ministre sur rapport du Ministre de l'Industrie après avis de la Commission Interministérielle des Installations Nucléaires de Base (CIINB) mise en place par le même décret, et avis conforme du Ministre de la Santé;

- l'autorisation de création fixe les caractéristiques de l'installation ainsi que les prescriptions particulières auxquelles doit se conformer l'exploitant, sans préjudice de l'application de la réglementation générale;
- la réglementation technique générale (RTG) concernant la sûreté des installations nucléaires de base est prise par arrêté du Ministre chargé de l'industrie.

Au sein du ministère de l'industrie, c'est le Service Central de Sûreté des Installations Nucléaires (SCSIN) qui est chargé des missions concernant la sûreté des installations nucléaires et en particulier de l'élaboration de la réglementation technique générale.

Ces dispositions ont été prises en raison de l'évolution encore rapide des technologies et des idées en matière de sûreté, qui font que la réglementation technique ne saurait actuellement dispenser de l'examen cas par cas des différentes installations pour juger si leur degré de sûreté est suffisant.

Dans ce cadre, le SCSIN fait procéder pour chaque installation à un examen technique de l'installation sur la base du rapport préliminaire de sûreté remis par l'exploitant. Cet examen technique est effectué par le Département de Sûreté Nucléaire (Institut de Protection et de Sûreté Nucléaire, CEA) qui rapporte les conclusions de son analyse devant un groupe d'experts (Groupe Permanent) nommé par le Ministre de l'Industrie.

Le Groupe Permanent, à l'issue de ses travaux, émet un avis assorti de propositions de prescriptions techniques.

Le décret d'autorisation de création est rédigé au vu des observations et propositions du Groupe Permanent. Les prescriptions du décret sont le plus souvent précisées ou complétées par des observations transmises à l'exploitant avec la notification du décret.

3. D'autres considérations éclairent les conditions dans lesquelles se développe la réglementation technique en France.

a) Les règlements fixent un certain nombre de règles de caractère essentiellement technique et sont considérés comme des conditions nécessaires, mais pas forcément suffisantes, à l'obtention d'un degré de sûreté satisfaisant. C'est aux constructeurs et exploitants qu'il appartient de prendre les mesures nécessaires pour que le degré de sûreté recherché soit atteint, et d'en apporter la preuve aux autorités de sûreté.

b) Il est nécessaire que la validité de ces règles ait été confirmée au plan de la sûreté par les analyses de sûreté, les essais de sûreté et l'expérience de fonctionnement.

c) Pour permettre aux constructeurs et exploitants le plein exercice de leurs responsabilités, et pour ne pas bloquer le développement d'une technologie encore susceptible d'améliorations, les règlements édictés en France visent davantage à fixer des objectifs qu'à imposer des moyens pour les atteindre, le choix des moyens étant laissé le plus souvent aux constructeurs et exploitants. Certaines dispositions peuvent cependant être exclues par la réglementation.

II - Etat d'avancement de la réglementation technique générale

1. Protection contre les dangers des rayonnements ionisants et rejets d'effluents radioactifs liquides et gazeux

Les règlements en la matière ne sont pas pris au titre de la réglementation technique générale en matière de sûreté nucléaire, mais ne peuvent être ignorés si l'on considère les objectifs de la sûreté. On rappelle seulement ci-dessous les textes principaux :

- Protection contre les dangers des rayonnements ionisants :

  . Décret 66-450 du 20 Juin 1966 relatif aux principes généraux de protection contre les rayonnements ionisants; ce décret fixe les équivalents de dose maximaux admissibles pour les travailleurs, les personnes non directement affectées à des travaux sous rayonnement, le public et la population dans son ensemble, et définit des principes généraux de protection et de surveillance.

  . Décret 75-306 du 28 Avril 1975 relatif à la protection des travailleurs contre les dangers des rayonnements ionisants dans les installations nucléaires de base; ce décret fixe les mesures d'ordre administratif, les mesures concernant la zone contrôlée et les mesures d'ordre médical concernant le personnel de la zone contrôlée.

On note que ces deux textes ont été rédigés en accord avec les recommandations de la Commission Internationale de Protection Radiologique. Leur mise à jour est prévue pour tenir compte des dernières recommandations de cette commission, notamment ses publications 22 et 26.

- Rejets d'effluents radioactifs liquides et gazeux :

  . Décret 74-945 du 6 Novembre 1974 relatif aux rejets d'effluents radioactifs gazeux.

  . Décret 74-1181 du 31 Décembre 1974 relatif aux rejets d'effluents radioactifs liquides.

Ces textes fixent les conditions dans lesquelles peuvent être effectués les rejets radioactifs liquides et gazeux, définissent les études qui doivent être présentées à l'administration pour obtenir les autorisations nécessaires, ainsi que les moyens de contrôle et de surveillance qui doivent être utilisés.

Les textes généraux ci-dessus prévoient qu'une autorisation fixant les limites de rejet doit être donnée pour chaque réacteur sous forme d'arrêté conjoint des Ministres intéressés, notamment le Ministre de l'Industrie et le Ministre de la Santé.

2. Réglementation technique générale concernant la sûreté des installations nucléaires de base comportant un réacteur

Un groupe de travail a été désigné par les autorités de sûreté (Service Central de Sûreté des Installations Nucléaires), pour définir le cadre dans lequel s'inscrirait la réglementation technique générale. Ce groupe a proposé que les textes soient répartis en trois niveaux hiérarchisés suivant leur degré de généralité :

1° niveau : Principes fondamentaux de sûreté

Un projet de texte a été rédigé par le groupe de travail précité. Il sert actuellement de base pour la rédaction des textes des autres niveaux.

Il porte sur la conception, la qualité, l'acceptation des risques, le choix des sites et les obligations des constructeurs et exploitants.

Il fixe notamment le principe général suivant :

" Il est nécessaire que les effets - notamment biologiques - des rayonnements ionisants émis par un réacteur nucléaire et les produits radioactifs qui s'y trouvent soient en permanence maitrisés. Des mesures permettant d'assurer le confinement des produits radioactifs et l'atténuation des rayonnements ionisants doivent être prises à cet effet lors de la conception, de la construction, de la mise en service, de l'exploitation et de l'arrêt définitif de l'installation".

En ce qui concerne le principe d'acceptation des risques, il est prévu que:

" En fonctionnement normal, les équivalents de doses susceptibles d'être reçus par les travailleurs de l'installation, les personnes du public et la population ne devront, en aucun cas, dépasser les limites admissibles fixées par les autorités compétentes. Les risques de conditions accidentelles conduisant à des effets supérieurs à ceux admis en fonctionnement normal devront être limités comme suit: une condition accidentelle devra être d'autant plus improbable que les dommages qui en résulteraient pourraient être graves, l'acceptation se faisant par référence aux risques de toutes sortes acceptés dans les activités courantes de la société".

2° niveau : Critères généraux de sûreté

Ces textes sont actuellement rédigés sous forme de projet et ont été examinés par le groupe de travail précité. Il y a quatre documents principaux:

- Critères généraux relatifs aux sites,
- Critères généraux relatifs à la conception des réacteurs nucléaires,
- Critères généraux relatifs à l'exploitation des réacteurs nucléaires,
- Critères généraux relatifs à l'assurance de la qualité.

Les critères généraux relatifs à la conception comprennent une partie générale applicable à toutes les filières de réacteur, et des parties applicables, soit aux réacteurs PWR, soit aux réacteurs à neutrons rapides.

Lorsqu'ils atteignent un niveau suffisant d'élaboration les projets servent de référence :

- d'une part au titre de la procédure d'autorisation des installations, dans les rapports d'évaluation du Département de Sûreté Nucléaire et dans les discussions du groupe d'experts technique précité (Groupe Permanent) placé auprès du SCSIN.
- d'autre part au titre de la surveillance des installations, par les inspecteurs des installations nucléaires de base.

Ceci constitue pour eux une épreuve d'utilisation expérimentale.

La procédure en vigueur prévoit également que ces textes seront soumis à l'examen technique du Groupe Permanent.

3° niveau : Prescriptions techniques particulières

Ces textes portent sur des systèmes particuliers ou sur des moyens de prévention contre des risques particuliers.

a) Textes publiés sous forme d'arrêtés.

- Caissons de réacteurs nucléaires en béton précontraint.

Un arrêté du 15 Juin 1970 relatif aux caissons de réacteurs nucléaires en béton précontraint, utilisés à l'époque pour les réacteurs nucléaires refroidis au gaz, fixe les objectifs de sûreté qui doivent être atteints, les choix techniques effectués par les constructeurs devant être justifiés. Cet arrêté comprend des recommandations relatives aux calculs, à la construction, aux essais et à la surveillance du caisson en exploitation.

- Chaudières nucléaires à eau sous pression.

Les chaudières nucléaires à eau constituent des appareils à pression et sont à ce titre soumises à une réglementation particulière qui s'ajoute à la réglementation plus générale relative à la sûreté nucléaire.

Cette réglementation comprend en particulier l'arrêté du 26 Février 1974 relatif au circuit primaire principal des chaudières nucléaires à eau sous pression fixant des objectifs de sûreté dans le domaine des calculs, de la construction, des épreuves et de la surveillance des circuits primaires principaux des réacteurs à eau.

Les Services Interdépartementaux de l'Industrie et des Mines sont chargés de contrôler l'application de la réglementation des appareils à pression, le Service de l'Industrie et des Mines de Bourgogne - Franche Comté étant chargé du contrôle de l'application de l'arrêté du 26 Février 1974 à la construction des chaudières nucléaires à eau sous pression (Bureau de Contrôle de la Construction Nucléaire - BCCN). En cas de difficulté dans l'application de la réglementation, les services de contrôle peuvent recourrir à la commission des appareils à pression qui a délégué à sa section permanente nucléaire le soin d'examiner les questions relatives à l'application de la réglementation des appareils à pressions aux chaudières nucléaires. Le cas échéant l'arbitrage du Ministre de l'Industrie peut être sollicité.

b) Textes rédigés sous forme de projet ou en cours de rédaction.

Ces textes portent sur le protection contre l'incendie, la prise en compte des séismes dans la sûreté des réacteurs, les systèmes de protection. Le développement d'autres textes est également envisagé.

4° niveau : Approbation de codes et normes par les autorités de sûreté

Comme on l'a vu, la réglementation technique générale des niveaux précédents s'applique principalement à fixer les objectifs de sûreté; il en résulte qu'il existe une solution de continuité entre la réglementation et les spécifications détaillées définissant les différents éléments d'une centrale nucléaire. L'utilisation de codes et normes permet de combler cette lacune. Compte tenu d'un système de révisions appropriées, les codes et normes peuvent être intégrés dans la réglementation technique générale, une fois approuvés par les autorités de sûreté, formant ainsi un 4° niveau dans la hiérarchie des textes réglementaires.

- Codes et normes des réacteurs à eau sous pression

A l'initiative des pouvoirs publics, les constructeurs et exploitants français ont entrepris de rassembler dans un recueil les règles de conception et de construction des centrales nucléaires à eau sous pression. Ce recueil comprend les six parties suivantes actuellement à des stades d'élaboration différents, cinq d'entre elles étant en cours d'exmamen par les autorités de sûreté :

Procédés : Ce recueil précise : les critères généraux appliqués à l'ensemble de l'installation; les critères de conception applicables aux différents systèmes; les règles d'installation des systèmes et de liaison entre la chaudière nucléaire et les autres parties de l'installation; les règles suivies pour les études de fonctionnement et celles relatives à la protection des personnes contre les rayonnements ionisants.

Matériel électrique : Ce recueil contient les règles d'étude, de fabrication de qualification et d'essais des matériels électriques.

Matériels mécaniques : Ce recueil comprend cinq tomes. Le premier renferme les dispositions générales portant notamment sur la conception, la fabrication, le contrôle, les épreuves et les essais des matériels. Les autres tomes renferment des règles correspondant à différents domaines techniques : approvisionnement, méthodes d'essais destructifs et non destructifs, les opérations de soudage et les autres opérations de fabrication.

Génie Civil : Ce recueil définit les critères de conception et de réalisation applicables au bâtiment réacteur et aux autres bâtiments de l'ilôt nucléaire, et précise les conditions d'épreuve d'enceinte de confinement.

Combustible : Pas encore disponible.

Protection contre l'incendie : Ce recueil définit les règles de conception particulièresen vue de la prévention et de la protection eu égard aux risques d'incendie dans les centrales nucléaires.

- Normes éditées par l'Association Française de Normalisation (AFNOR)

Des procédures en place depuis de nombreuses années permettent l'élaboration de normes nationales dans les domaines touchant les réacteurs nucléaires. Certaines de ces normes peuvent être rendues d'application obligatoire pour la sûreté lorsqu'elles sont citées comme telles dans un arrêté pris au titre de la RTG.

III - Utilisation de la réglementation technique dans la procédure d'autorisation

1. Réglementation technique générale

Comme on l'a indiqué plus haut, la procédure d'autorisation s'accompagne d'un examen technique de chaque installation nucléaire de base à créer. Cet examen technique repose sur les éléments principaux suivants :

- rédaction d'un rapport de sûreté par l'exploitant futur de l'installation (EdF pour les réacteurs électronucléaires);
- évaluation technique du rapport de sûreté par le Département de Sûreté Nucléaire (IPSN - CEA);
- examen du rapport de sûreté par le Groupe Permanent précité sur la base du rapport d'évaluation du DSN et émission d'un avis par ce groupe à l'intention du SCSIN.

Les différents examens techniques ci-dessus sont poussés au-delà d'une simple vérification de conformité; ils s'appuient sur la réglementation technique générale et sur d'autres documents de référence en tant que de besoin.

2. Autres documents utilisés comme référence technique

a) Le Chef du SCSIN adresse, le cas échéant, des directives aux exploitants notamment dans le cadre de la définition d'options de sûreté valables pour une série de réacteurs, options préalablement examinées par le Groupe Permanent. De telles directives servent bien évidemment de base aux évaluations de sûreté des réacteurs concernés.

b) Comme cela a été rappelé ci-dessus, le Groupe Permanent formule des avis, adressés au SCSIN, en conclusion des examens qu'il effectue. Ces avis ou recommandations du Groupe Permanent sont utilisés ensuite pour l'analyse de la sûreté des réacteurs similaires.

c) Pour les réacteurs surrégénérateurs, des recommandations relatives aux critères de sûreté pour la centrale de 1200 MWe à Neutrons Rapides Super Phénix ont servi de base aux examens techniques de sûreté après leur approbation par les autorités de sûreté.

d) Les principes fondamentaux de sûreté (niveau 1 de la RTG) introduisent le concept de risque et permettent d'envisager pour l'avenir le développement de critères de caractère probabiliste.

Les difficultés rencontrées dans ce domaine sont bien connues et le constat de la situation française a déjà été dressé [17].

Toutefois dans certains domaines comme la prise en compte des chutes d'avions, des critères probabilistes provisoires sont pris en considération dans les analyses de sûreté.

e) Pour les réacteurs à eau sous pression, il peut être fait appel en tant que de besoin à l'examen de références étrangères pour des problèmes d'appréciation difficile.

IV - Conclusions

La réglementation technique générale en matière de sûreté a été développée en France plus particulièrement dans les domaines où le développement de l'énergie nucléaire a bénéficié d'un acquis antérieur comme c'est le cas du circuit primaire principal des réacteurs à eau sous pression avec la réglementation sur les appareils à pression.

Parallèlement un effort a été entrepris sous l'impulsion des pouvoirs publics pour développer des codes et normes plus détaillés permettant de préciser les règles s'appliquant aux différents systèmes et matériels constituant les centrales nucléaires.

Cette situation doit être appréhendée dans le cadre des pratiques françaises en matière de sûreté dont deux aspects essentiels sont :

- la procédure d'autorisation très stricte qui repose sur un examen cas par cas des installations à divers stades de leur développement,
- la technique d'analyse de sûreté par barrière qui a déjà fait l'objet de plusieurs publications /2/.

Pour l'avenir on peut s'attendre à ce que l'effort entrepris soit poursuivi, les autorités de sûreté, le CEA et EdF, portant par ailleurs un grand intérêt et apportant leur participation effective aux efforts menés au plan international pour le développement de recommandations en matière de sûreté nucléaire.

BIBLIOGRAPHIE

/1/ - P. TANGUY et F. COGNÉ - Utilisation des méthodes probabilistes dans l'évaluation de sûreté des installations nucléaires. NUCLEX - (Bâle, 3-7 October 1978)

/2/ - J. BOURGEOIS - La sûreté des Installations Nucléaires - Guide International de l'Energie Nucléaire. Edition LESOURD, 1978.

GERMAN RULES AND REGULATIONS WITH SPECIAL REFERENCE TO APPLICATION DOCUMENTS

A. Kraut
Gesellschaft für Reaktorsicherheit (GRS) mbH
Cologne, Federal Republic of Germany

Regeln und Richtlinien spielen eine wesentliche Rolle bei der Entwicklung einer sicheren und ökonomischen Technologie sowie bei der systematischen, effektiven und transparenten Durchführung von Genehmigungsverfahren. Die auf dem Gebiet der Kerntechnik zur Anwendung kommenden deutschen Vorschriften werden hier mit Bezug auf die regelerarbeitenden Organisationen vorgestellt. Detaillierter wird auf die Vorschriften eingegangen, in denen Anforderungen an erforderliche Unterlagen im atomrechtlichen Genehmigungsverfahren gestellt werden.

Regulations and standards play an essential role in achieving a safe and economic technology and in making the licensing procedure systematic, effective and clear. German rules and regulations applicable to the nuclear field are presented in this paper together with references to the rulemaking organizations. Detailed information is given on those rules and regulations, which prescribe the requirements concerning necessary documents for the nuclear licensing procedure.

1. Introduction

Rules and regulations play an essential role in achieving a safe and economic technology and in making the licensing procedure systematic, effective, and clear. In Germany there is a plurality of different rules and regulations used in the nuclear field. Therefore, the order of those rules and regulations is elucidated in fig. 1.

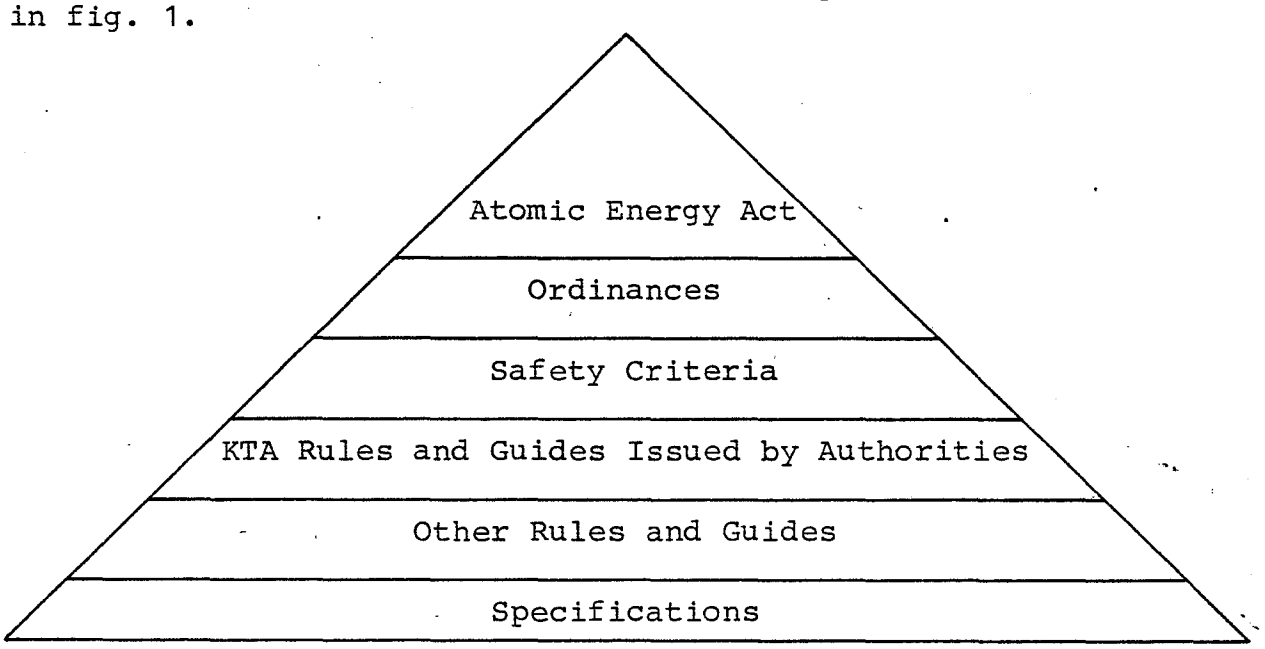


Fig. 1 - Nuclear Engineering Regulations

2. German Regulations

2.1 Atomic Energy Act and Associated Ordinances

The legal basis for the peaceful utilization of nuclear energy was created in 1959 by an amendment to the Basic Law and the promulgation of the Atomic Energy Act [1]. The Atomic Energy Act contains the basic provisions governing the peaceful utilization of nuclear energy. More extensive legal provisions are contained in a number of ordinances issued on the basis of that Act. Apart from the existing ordinances such as Radiological Protection Ordinance [2], Nuclear Licensing Procedures Ordinance [3] or Nuclear Financial Security Ordinance [4], a total of 10 further ordinances are in the planning stage at present [5].

2.2 Safety Criteria

The concretization of the protective aims set forth in the Atomic Energy Act and the associated ordinances requires technical regulations. The major safety aspects are contained in the Safety Criteria for Nuclear Power Plants [6]. After having been presented for comment to all those involved in the nuclear licensing process, these Safety Criteria were approved by the Federal States Committee for Nuclear Energy. The Safety Criteria have been developed in particular for LWR nuclear power plants.

However, they do also apply in their entirety to the non-plant-specific requirements and by analogy to the plant-specific requirements of all other types of nuclear power plants.

2.3 KTA (Nuclear Engineering Committee) Rules and Guides Issued by the Authorities

As a matter of principle, all available rules pertaining to all relevant lines of engineering can be used for the further concretization of the licensing conditions and the requirements set forth in the Safety Criteria. However, the safety design of nuclear facilities on the basis of non-nuclear engineering principles is not always considered a sufficient approach. Thus, there is a genuine need for the preparation of a specific set of engineering rules and codes for the field of reactor safety.

The Nuclear Engineering Committee (KTA) is the institution to attend to the preparation and further the application of safety rules in those areas of nuclear engineering where as a result of experience a uniform view emerges among the experts of manufacturers, constructors and licensees of nuclear facilities as well as the reviewing organizations and authorities.

When preparing a set of rules or a code, the Nuclear Engineering Committee will draw on the work already performed by and the current assistance of other institutions, in particular the various technical committees within the German Institute for Standardization (DIN) and the Pressure Vessel Association (AD). The Nuclear Engineering Committee (KTA) is to use its best efforts to achieve a canalization of the various initiatives in the field of the preparation of safety rules for nuclear engineering.

The KTA Manual [7] contains information on the KTA Rules and their present status

Although a canalization of the preparation of rules and codes towards the Nuclear Engineering Committee (KTA) is attempted, the Committee is nevertheless not the only institution to set up the safety standards for the safety-related assessment of nuclear power plants. The authorities concerned have to reserve the right to set up safety-related standards in individual cases where the preconditions for the preparation of safety-related rules as required in accordance with the KTA Bulletin [8] have not yet been met.

Thus, there are, apart from private initiatives and outside the frame of the Safety Criteria and KTA Rules, guides and guidelines on reactor safety aspects which are developed under the auspices of the authorities. The motive for the preparation of such guides and guidelines is, among other things, that the authorities are obliged to make a decision on a given licensing application or some supervisory action and, prior to making that decision, have to clarify the organizational, administrative, engineering and safety-related aspects involved, if necessary under consultation of experts.

The Manual on Reactor Safety and Radiological Protection [9] pub-

lished by the Federal Ministry of the Interior (BMI) contains information on the guides prepared by the authorities. In addition, the Manual contains the guidelines and recommendations issued by the commissions advising the Federal Ministry of the Interior, i.e. Reactor Safety Commission (RSK) and Radiological Protection Commission (SSK).

2.4 Other Rules and Guides

Nuclear rules are also developed and/or prepared by private initiative. This is done either through a joint effort of different interest groups or through the activities of individual groups. Although the criterion for membership in standardization committees is not a parity of interests but technical qualification, the standards [10] issued by the Standardization Committees on Nuclear Engineering and on Radiobiology are examples of cooperation among different groups. On the other hand, the directives [11] and decisions of the TÜV Coordination Office for Nuclear Engineering within the frame of the Association of Technical Supervisory Inspectorates (VdTÜV) will have an impact on national rules and codes even if they are primarily intended as a tool for the regulation of the efforts of the Technical Supervisory Inspectorates and of the Gesellschaft für Reaktorsicherheit (GRS).

Moreover, a number of institutions that prepare and issue rules pertaining to conventional engineering have also turned to the preparation of nuclear engineering rules of their own. Thus, a certain number of other nuclear rules exist beside the KTA rules and the guides issued by the authorities. Where no relevant KTA rules, guides issued by the authorities or other nuclear rules are available, safety-related guides, standards and recommendations relating to non-nuclear engineering will also be applied following a relevant review.

As far as nuclear facilities are concerned, the following major rules and guides are taken into consideration beside the DIN Standards [10]:

- Reference Sheets issued by the Pressure Vessel Association [12]
- Engineering Rules for Steam Boilers (TRD) [13]
- Rules for the Prevention of Industrial Accidents issued by the Workmen's Compensation Insurance Cooperatives [14]
- VDE Regulations [15]
- VDI Guides [16]
- VdTÜV Reference Sheets [17]
- Materials Reference Sheets and other documents issued by the Association of German Metallurgists [18].

Some of the listed rules and guidelines, especially some of the guidelines approved by the authorities, deal directly with the standardization and systematization of the licensing procedure in that they give instructions on the format and content of application and licensing documents.

3. Application Documents

The procedure for licensing the construction and operation of nuclear power plants, which is required under Section 7 of the Atomic Energy Act, is laid down in the Nuclear Licensing Ordinance. Section 2 of this ordinance prescribes the form and content of the application, Section 3 specifies the documents, which are necessary for the examination of the licensing prerequisites.

The following documents are required in particular

- a safety analysis report,
- supplementary plans, drawings and descriptions of the plant and its components,
- information on the physical security of the plant,
- information on the qualification of personnel,
- safety specifications,
- suggestions for the provisions of financial security,
- information on the noncontamination of water, air and soil.

3.1 Safety Analysis Report

According to the Nuclear Licensing Ordinance the Safety Analysis Report shall contain

- a description of the projected plant and its operation, supported by corresponding maps, location and layout plans,
- a description of the impacts and hazards involved in the plant and its operation, and
- a description of the precautions to be taken in accordance with the state of the art.

Further details for the drafting of safety analysis reports are given by the "Checklist for a Standard Safety Analysis Report for Nuclear Power Plants with Pressurized Water Reactor or Boiling Water Reactor". This checklist, which has been prepared by a working group convened by the Federal Minister of the Interior, prescribes the following aspects to be dealt with in safety analysis reports

- Site
- Power plant
- Radioactive materials and radiological protection measures
- Operation of the power plant
- Accident analysis

- Decommissioning

Beside detailed items concerning each chapter the checklist presents a list of accidents that have to be regarded.

3.2 Compilations of Required Information

The German licensing procedure provides for a number of partial construction permits the first of which will refer to site and concept of a plant. Design, erection, manufacturing and other details are discussed in later partial construction permits. This is why the applicant will generally submit detailed documents at a later time. However, the entirety of the information to be submitted is not laid down in the Checklist for a safety analysis report but in a number of guidelines which are prepared by the GRS in cooperation with the Technical Supervisory Associations. Industrial organizations are also called in at a later time before the guidelines are approved by the States Committee on Nuclear Energy.

These "Compilations of the information that is required for examination purposes in licensing procedures for nuclear power plants under the Atomic Energy Act" do not only contain the necessary information but also allocate the information to certain hold points at which it has to be available at the latest. However, these hold points are not the individual licensing steps as such but the individual phases in the construction of the plant or of plant components.

Examples for such hold points are:

- Construction of buildings
- Manufacture of semi-finished products
- Manufacture of components
- Erection of systems
- Commissioning of systems
- Commissioning of the plant

The orginally 29 papers, for example on site, reactor core, reactor pressure vessel etc, are presently being condensed to approximately 18 comprehensive guidelines.

3.3 Operating Manual and Safety Specifications

Instruction concerning formation and content of the operating manual are given by the rule 1201 of the Nuclear Engineering Committee "Requirements for the Operating Manual". Within the 4 main parts

- operating regulations

- operation of the plant
- incidents
- operation of systems

the operating manual shall contain all valid operating regulations and all operational and safety relevant instructions for the shift personnel, which are necessary for operation and for coping with incidents.

The safety specifications, which are intended to warrant a rapid and complete proof of all data, limits and measures of importance to the safety of the nuclear power plant and its operation, are also to be integrated into the operating manual. A seperate Checklist for Format and Content of Safety Specifications for Nuclear Power Plants" is given in the "Guidelines Concerning the Requirements for Safety Specifications for Nuclear Power Plants".

The following major subjects shall be covered by the safety specifications among others:

- Organization of the operating staff
- all safety-related requirements for the operation of the plant issued by the authorities
- all safety-related operational restrictions in case of missing redundancy including the permissible outage and repair time of the partial systems or components concerned
- all events the occurrence of which necessitates a report to the supervisory authority
- all protective and hazard limits including the reactor protection values and hazard alarms which
  - necessitate a power restriction for reasons of safety (e.g. redundancy failure) or
  - serve the protection of the persons who are in the nuclear power plant (e.g. activity limits of loop monitors, area air monitors and local dose rates) or
  - indicate an impermissible environmental exposure (e.g. activity limits in exhaust air and liquid wastes)
- flow charts for incidents and accidents including the necessary manual intervention as well as the measures which are taken automatically

4. Summary

To arrive at a rationalization of the licensing procedures the information required for examination purposes shall be defined. The documents to be submitted, such as the safety analysis report, shall be standardized to the extent possible. The rules and guidelines, which have been introduced in this context, serve this purpose.

Bibliographic References

[1] Gesetz über die friedliche Verwendung der Kernenergie und den Schutz gegen ihre Gefahren (Atomgesetz) vom 23. Dezember 1959 (BGBl. I, S. 814), in der Fassung der Neubekanntmachung vom 31. Oktober 1976

[2] Verordnung über den Schutz vor Schäden durch ionisierende Strahlen (Strahlenschutzverordnung - StrlSchV) vom 13. Oktober 1976 (BGBl. I, S. 2905)

[3] Verordnung über das Verfahren bei der Genehmigung von Anlagen nach § 7 des Atomgesetzes (Atomrechtliche Verfahrensverordnung - AtVfV) vom 18. Februar 1977 (BGBl. I, S. 280)

[4] Verordnung über die Deckungsvorsorge nach dem Atomgesetz (Atomrechtliche Deckungsvorsorgeverordnung - AtDeckV) vom 25. Januar 1977 (BGBl. I, S. 220)

[5] Zur friedlichen Nutzung der Kernenergie; Dokumentation der Bundesregierung, 1977, S. 368, Hrsg.: Das Bundesministerium für Forschung und Technologie, Bonn-Bad Godesberg

[6] Sicherheitskriterien für Kernkraftwerke, verabschiedet im Länderausschuß für Atomkernenergie am 12. Oktober 1977, Bundesanzeiger Nr. 206 vom 3.11.1977

[7] KTA-Handbuch, Geschäftsstelle des Kerntechnischen Ausschusses bei der Gesellschaft für Reaktorsicherheit mbH, Köln

[8] Bekanntmachung über die Bildung eines Kerntechnischen Ausschusses; Fassung vom 27. September 1974, Bundesanzeiger Nr. 193 vom 15.10.1974

[9] Handbuch der Reaktorsicherheit und des Strahlenschutzes, 1978; Hrsg.: Der Bundesminister des Innern, Bonn

[10] DIN-Verzeichnis der Normen und Norm-Entwürfe, Beuth Verlag GmbH, Berlin

[11] Weisungsbeschlüsse der TÜV-Leitstelle Kerntechnik bei der VdTÜV, Maximilian Verlag, Herford

[12] AD-Merkblätter, Carl Heymanns Verlag KG, Köln

[13] Dampfkesselbestimmungen (TRD), Carl Heymanns Verlag KG, Köln

[14] Unfallverhütungsvorschriften (UFF), Carl Heymanns Verlag KG, Köln

[15] VDE-Bestimmungen, VDE-Verlag GmbH, Berlin

[16] VDI-Richtlinien, Beuth Vertrieb GmbH, Köln

[17] VdTÜV-Merkblätter, Maximilian-Verlag, Herford

[18] VDEh-Werkstoffblätter, Verlag Stahleisen mbH, Düsseldorf

[19] Regeln und Richtlinien für die Kerntechnik - Mainz 1977, Hrsg.: Deutsches Atomforum e.V., Bonn

[20] Fichtner, N., K. Becker und M. Bachir: Catalogue and classification of technical safety standards, rules and regulations for nuclear power reactors and nuclear fuel cycle facilities, EUR 5849, Commission of the European Communities, Luxembourg, 1977

First Period of Disscussions on Session III

J. van Daatselaar (Netherlands) (To J.P. Pelé)

With respect to harmonization in the paper is the statement that there is no reason to allow the development of different overall levels of safety to which individuals of different nationalities are exposed.

I think that there could be a number of reasons to allow the development of different safety levels for individuals in different countries:

(1) Even if the risk to individuals is the same, the risk to the population as a whole might differ, depending upon the number of people exposed. If we think that the risk to the population has to be the same, then the individual risks will differ.

(2) A second reason might be the measure in which nuclear power is needed. If there are no or only a few other sources for energy production, I think you have to accept a higher risk from one nation to another.

(3) Another reason, from quite a different approach is the following: if the aim is to harmonize the design requirements or the construction of nuclear power plants to avoid trade hindrances, power plants of the same vendor will have the same release-probability curve. But then the risk to individuals or to the population depends upon the population distribution around the site and will be different for different nations. Requiring the risks to be the same, would require differences in design. Would you be so kind as to comment on these facts or to clarify your statement.

J.P. Pélé (CEC)

(1) Le risque est perçu dans le public surtout au niveau de l'individu.

(2) Il est possible que certaines nations soient prêtes à admettre des risques plus grand pour leurs ressortissants mais (a) ce n'est pas une raison suffisante pour le faire accepter au niveau de la Communauté; et (b) on ne peut pas imposer de plus grands risques à des populations voisines d'autres états membres.

(3) Il ne faut pas confondre un essai d'harmonisation des exigences règlementaires ou de normes industrielles avec une standardisation du matériel.

S. Israel (US) (To J.P. Pelé)

In developing seismic criteria, will your organisation consider the potential failure of non-safety grade equipment (mainly in

in the auxiliary building) when addressing the requirement to shut down the reactor and maintain safe shutdown?

J.P. Pélé (CEC)
Comme il a été indiqué, les 'effets sismiques' sont un des sujets étudiés dans le cadre du groupe de travail et pour lequel un rapport de synthèse, comparant les différents pratiques nationales a été preparé, mettant en lumière les points de convergences et les points de divergences; il ne s'agit pas, pour l'instant, du développement de critères propres.

S.Israel (US) (To A. Kraut and P. Lebouleux)
(1) How long does it take to develop a safety guide?
(2) While the guide is being developed, how are the requirements addressed in current plant reviews?
(3) Do you have provisions for instantly imposing requirements on operating plants and plants under review?

A. Kraut (F.R. of Germany)
(1) On average, 3 to 4 years.
(2) By the requirements given in the partial construction permits or the addenda to these permits. The requirements are mainly based on the recommendations given in the safety assessments.
(3) According to Paragraph 17 of the Atomic Energy Act, a license can be cancelled if the nuclear facility may endanger the staff and/or the general public. Backfitting measures may be required in this context.

P. Lebouleux (France)
La rédaction d'un guide de sûreté s'effectue à la suite d'un long processus d'étude, de reflection, d'essais, dont les conclusions peuvent faire l'objet d'un rapport technique préliminaire. J'ai l'expérience personnelle de la rédaction d'un guide sur la protection contre l'incendie qui a duré 1 an. Ensuite, la transformation du projet ainsi obtenu peut durer encore longtemps avant sa mise en application sur le plan réglementaire, car le projet est d'abord mis à l'essai, comme je l'ai indiqué dans ma communication.
Lorsqu'il y a des problèmes à régler rapidement - et que l'on ne dispose pas de normes ou de règles satisfaisantes - on utilise la procédure qui a été décrite dans la communication: analyse technique du problème par le DSN et éventuellement par le Groupe Permanent, pour décision par le SCSIN. Même s'il y avait des règlements disponibles on utiliserait cette procédure et on examinerait, parallèlement aux directives données à l'expoitant, si le règlement lui-même ne doit pas être révisé compte tenu de l'enseignement tiré de l'éxamen technique effectué.

J. Edwards (UK) (To A. Kraut and P. Lebouleux)
Could the authors of the three papers say how their countries would interpret the phrase 'as low as reasonably achievable' and whether they anticipate any problems in this field. Would not differences perhaps arise between the interpretation placed on this by the labour unions and by the owner or regulatory body?

A. Kraut (F.R. of Germany)
A final say concerning this problem should be with the health physicists and the politicians. Difficulties may arise as long as there does not exist a clear cost/benefit basis.

P. Lebouleux (France)
La réglementation française pour la protection des travailleurs dans les installations nucléaires de base (INB) prend en compte un concept voisin de ALARA en demandant que les doses individuelles soient aussi faibles que possible et que le nombre de personnes soumises à des irradiations soit également aussi faible que possible. Cependent ALARA ne peut être quantifié de façon générale puisqu'il s'agit d'une analyse coût/bénéfice à effectuer dans chaque cas particulier. En France la protection des travailleurs est sous le contrôle du Ministère du Travail (code du travail) et du Ministère de la Santé. Dans tous les établissements il existe des commissions d'hygiène et de sécurité dans lesquels les représentants du personnel ont des sièges. Les problèmes liés aux irradiations des personnes sont discutées dans ces instances.

N. Aybers (Turkey) (To A. Kraut)
On page 5 of your paper there is mention of certain hold points. I did not understand what you mean by manufacture of semi-finished products and also manufacture of components. These might be some major components, it is not possible to have as a hold point all the components to be manufactured. Could you please explain more about these hold points?

A. Kraut (F.R. of Germany)
During the manufacturing process for a special system, e.g. the pressure suppression system, some information must be given at the early stages, e.g. statements on materials or statements on postulated loads before the manufacture of semi-finished products or statements concerning welding procedures and demonstration of adequate and valid process test before the manufacture of relevant components.

SAFETY GUIDES DEVELOPMENT PROCESS IN SPAIN

M. Perelló, J.L. Butragueño
and
J. Reig
Junta de Energía Nuclear

MADRID, Spain

Las guías de seguridad se han convertido en uno de los factores más importantes en su aplicación al proceso de autorización de instalaciones del ciclo del combustible, incluidas las centrales de potencia.

A medida que la experiencia corrobora cada vez mejor los métodos y procedimientos de ingeniería, sus resultados se recogen en forma de guías, normas y documentos análogos.

Este trabajo presenta la experiencia actual en España sobre el desarrollo de guías y normas de seguridad. Se describe su proceso de elaboración y se incluyen comentarios sobre el futuro de las normas en las evaluaciones de seguridad.

Safety guides have become a major factor in the licensing process of nuclear power plants and related nuclear facilities of the fuel cycle. As far as the experience corroborates better and better engineering methodologies and procedures, the results of these are settled down in form of standards, guides, and similar issues.

This paper presents the actual Spanish experience in nuclear standards and safety guides development. The process to develop a standard or safety guide is shown.

Up to date list of issued and on development nuclear safety guides is included and comments on the future role of nuclear standards in the licensing process are made.

1. INTRODUCCION

Later on the decade of sixties, the Nuclear Safety Department was formed by just a few members, which should take over the revision of all the safety assesment concerning the construction - and initial operation of the first Spanish nuclear power plants. Most of the effort there was dedicated to assure that the plant was designed, erected and operated following the standards and safety guides published by the vendor countries, specially the US.

With the increase of the nuclear energy programme, the Nuclear Safety Department began to grow and was reorganized in three - Departments: Evaluation, Inspection and Radiological Protection.

The important role played by nuclear standards and its application to the licensing review recommended to create a Nuclear - Standards Service. This was done on September 1.974. The missions assigned to this Service are explained in section 2.

2. MISSIONS

The missions assigned to the Nuclear Safety Department of the J.E.N. (1) include among others, the followings:

1) To develop and to recommend nuclear safety standards, criteria and guides for siting, design, construction, generation and operation of reactors and other production and utilization facilities subject to the licensing.

2) To develop and recommend rules, regulation, standards and amendments there to for (a) the protection of the public from the possession, use, transfer and disposal of source, special nuclear and by-products material and from the generation of nuclear facilities - subject to licenses for operator, supervisor and the title for the radiological protection service head.

3) To coordinate J.E.N. participation in standards related activities of international and other national regulatory bodies, mainly those with which there is a colaboration agreement, such as the American NRC and the French CEA.

These missions can be sumarized in these points:

1. Develop safety guides based on current licensing process and on new practices as these practices are develop because of improvement in the state-of-the-art.
2. Work with others organizationes in developing national standard which are of value to the regultory practice.
3. Develop regulation and needed changes to regulation.

These three points can be considered as the objectives assigned to a special group of expertises which from an administrative point of view made theirs jobs actually in the Standards Service, inside the DSN. A new reorganization has been proposed and is under study by the J.E.N. Director to increase the administrative level and the human resources available today to deal with this task.

(1) Misiones, Actividades, y Desarrollo del Departamento de Seguridad Nuclear. DSN/II/11/75. Rev. Mayo 1.977.

3. THE PROCESS IN ACTION

For a better understanding of the process which will be described afterwards, it is important to remark two characteristics about the guides that are developed by the Nuclear Standards Service.

Firstly, the topics of the standards are essentially administrative procedures necessary to implement the Spanish nuclear regulations. This peculiarity is common to all newly formed standard developing organisms.

In addition, to draw an elaboration process it is most important to consider the mandatory level which the standard will have. The level will condition the people forming the working group.

Nowadays the safety guides which have been developed by the Nuclear Standards Service are only recommendations for better implementing the Nuclear Installations Act. Sometimes the fulfillmet of a safety guide is included in the conditions of the preliminary or construction permit; in this case the recommendation becomes a requirement.

The process described hereafter refers only to the writing stage in the life of the standard.

The writing stage begins with the discernment of a need by a potential user, usually the regulatory body.

A working group (W.G.) is nominated and a task leader is pointed out between the members of the Standards Service.

Standards working groups, composed of competent technical people, are the key element in the elaboration of nuclear standards.

The task leader should write the rough initial working paper, which would be presented to the W.G. in order to elaborate the first draft. This draft is disseminated through the others offices inside the Department and comments are collected. Those of these which seem appropiated are included into the text and a new version is written. Based on the comments received, succesive drafts are prepared.

After that, the paper comes over the Standards Service to rewrite it again in a formal standarized form. Later on, the paper is presented to the Direction of the Department for approval. If this is the case, the draft is printed and sent to others affected Govermental Offices and people, in general, with interest in the matter. The guide is used since this moment in the licensing process, at least, as the initial staff technical position.

With a year of periodicity, the guide is revised and all the public comments are solved. Those appropiated are included and the new revision is issued again.

If the problem is important enough from the safety point of view, the guide is proposed to the authority to be endorsed and published as a document with a higher mandatory level.

The whole process is scheduled for 45 weeks, a working-year.

In the initial steps, a task force of 4 to 5 experts is considered and, later on, the task leader is suposed to be able to push the job ahead by himself. Of course, some inscheduled informal meetings can take place to duscuss minor details.

The process described is general and can be changed according to other criteria, specially the temporary evolution. The time required for the writing varies with intensity of the need, the complexity of the subject, the state-of-the-art, and the management ability of those involved.

4. ACTUAL EXPERIENCE AND RELATED PROBLEMS

So far the Nuclear Standards Service has published a total of 11 safety guides and several other are in the process of its - approval. Enclosed you will find a complete list of these guides.

It is important to make clear that the nuclear industry has wellcomed our standards. Everything that clarifies what the regulatory body requires reduces licensing time. We have received plenty of comments from the utilities which will improve the standards, since the broader the collaboration is the better the final document becomes.

However there are many problems along the life of a standard. Those problems could be classified into two generic groups: internal problems, refered as those within the standard developing body, and external problems.

In Spain the main internal problem is common to all safety aspects: evaluation, inspection and regulation, that is the lack of people involved in licensing review. In the specific job of developing nuclear safety standards, this problem becomes hard to overcome. It is quite apparent that writing, coordinating, managing, interpreting, publishing, and maintaining up-to-date nuclear standards needs people, senior staff people with a good profesional background, necessary to produce requirements and guides. Also it is important that these people work fulltime in developing standards. Many standards projects appear to be taking much longer than necessary to accomplish. Several factors contribute to this, but the main one is that people has to do several tasks at the same time, of the most different nature.

Another major problem, specific of the Spanish situation, is a lack of well-defined objectives.

In a country like Spain which do not have a top-rank nuclear technology it is necessary to define the most convenient goal and lay out a plan and schedule for achievement. The present nuclear standards have been produced because of an identified need by the regulatory body, but there has been no serious analysis as to how the standards fit into the total need of the regulatory body and should set forth acceptable criteria for accomplishing necessary requirements and identify acceptable methods for meeting the criteria. In Spain these acceptable criteria and methods have been taken from other countries, more developed in nuclear technology. An obvious goal should be then, a set of criteria for implementing these "imported" standards, to the Spanish nuclear plants.

We have mentioned before that the utilities have sent comments to the published standards, but there is no doubt that collaboration in the early stage of developing a standard is the main.

external problem in Spain. There is an urgent need of people with experience in order to get a satisfactory draft standard to begin with.

5. CONCLUSIONS

- It is quite evident that the availability and use of good standards will substantially reduce licensing review time and simultaneously improve safety, reliability, and availability of nuclear plants. This implies the existence of a specific organism in charge of developing and adapting those standards.
- This body should have sufficient human and technical means to accomplish its task in a satisfactory way.
- In countries like Spain without their own nuclear design and technology, the objectives of developing nuclear standards should be oriented towards control and administrative guides, and criteria for implementing and adapting foreign standards. This will avoid innecessary duplication.
- We can continue as we have been with a lot of individual effort or we can increase the efficiency by working as a team. The efforts of the various regulatory bodies in the development of standards should be coordinated by a supervisory board of the CSNI, which will define common needs and fix adequate goals to cover them. That could lead us to a set of CSNI standards developed by all member countries.

LIST OF NUCLEAR SAFETY GUIDES PUBLISHED BY THE JUNTA DE ENERGIA NUCLEAR

(STANDARDS SERVICE OF THE SAFETY DEPARTMENT)

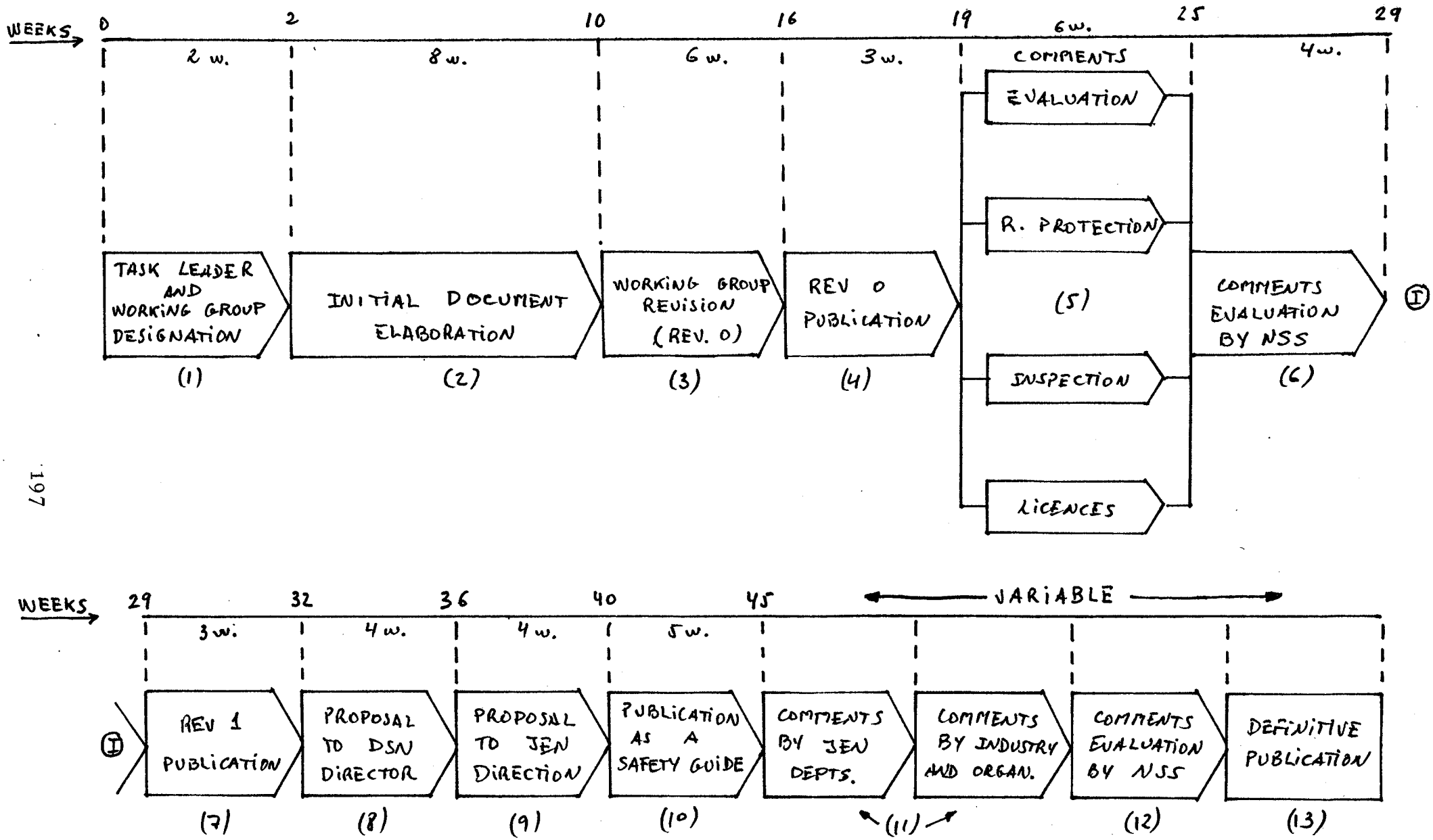
REF:	SPONSOR	TITLE	ESTADO
GSN-01/76	JEN	GUIA PARA SOLICITAR LA PUESTA EN MARCHA DE LAS INSTALACIONES DE MANIPULACION O ALMACENAMIENTO DE ISOTOPOS RADIACTIVOS (2ª y 3ª CATEGORIA) (COMISSIONING APPLICATION OF RADIACTIVE INSTALLATIONS)	publicada Junio/76
GSN-02/76	JEN	CUALIFICACIONES Y REQUISITOS EXIGIDOS A LOS CANDIDATOS A LA OBTENCION Y USO DE LICENCIAS DE OPERACION DE CENTRALES NUCLEARES DE POTENCIA. (REQUIREMENTS FOR OPERATORS OF NUCLEAR POWER PLANT)	publicada Sept/76
GSN-03/76	JEN	GUIA PARA EL ESTABLECIMIENTO DE UN PROGRAMA DE VIGILANCIA RADIOLOGICA AMBIENTAL EN LAS ZONAS DE INFLUENCIA DE LAS CENTRALES NUCLEARES. (ENVIROMENTAL MONITORING PROGRAMME)	publicada Oct./76
GSN-04/77	JEN	GUIA PARA LA OBTENCION DEL TITULO DE JEFE DEL SERVICIO DE PROTECCION CONTRA LAS RADIACIONES. (REQUIREMENTS FOR THE RADIATION PROTECTION MANAGER)	publicada Abril/77

.../...

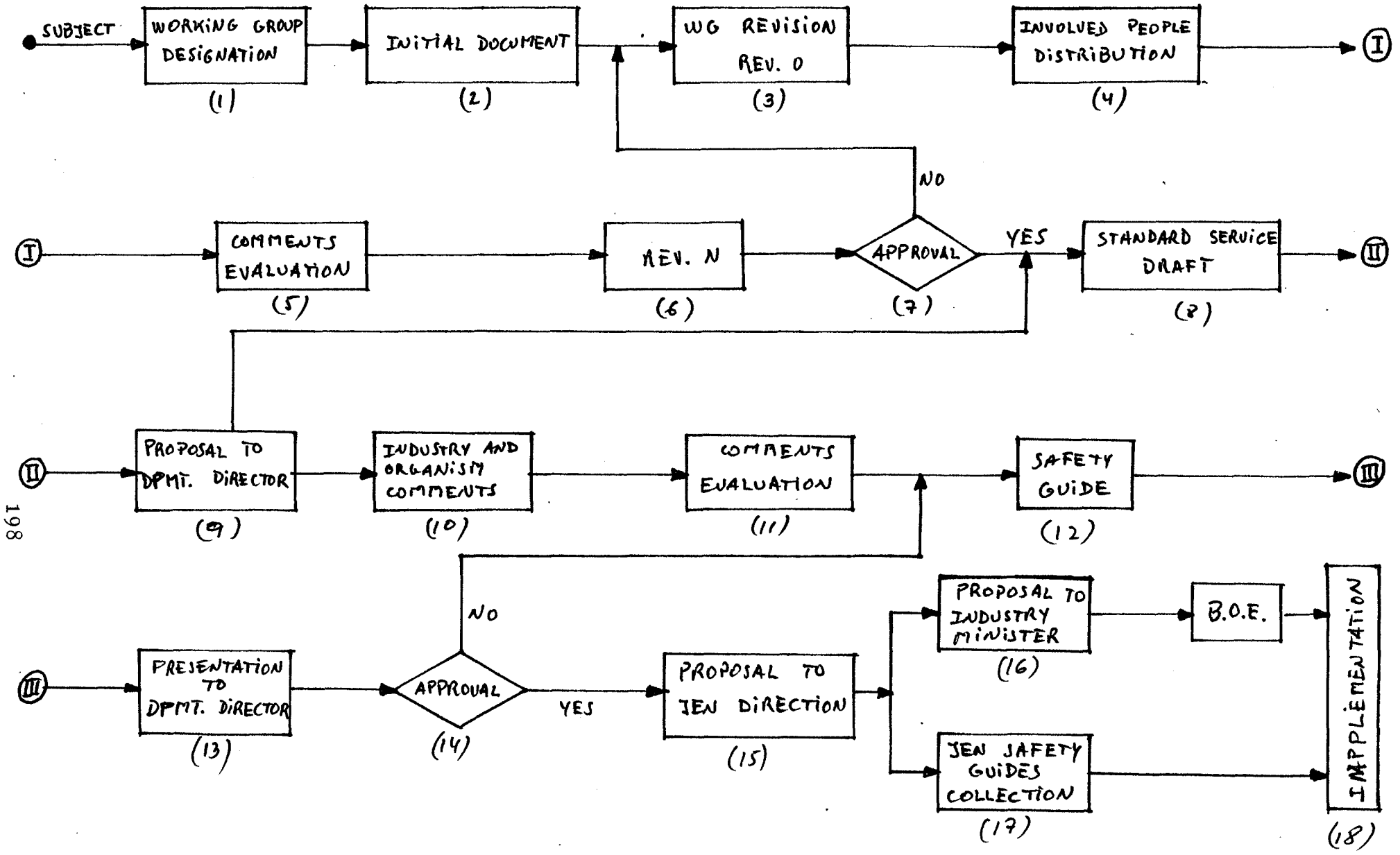
REF:	SPONSOR	TITLE	ESTADO
GSN-05/77	JEN	REQUISITOS FISICO PSIQUICOS EXIGIDOS A LOS CANDIDATOS PARA LA OBTENCION Y USO DE LICENCIAS DE OPERADORES Y SUPERVISORES DE INSTALACIONES NUCLEARES Y RADIACTIVAS. (REQUIREMENTS FOR SUPERVISORS OF NUCLEAR AND RADIOACTIVE FACILITIES)	publicada Julio/77
GSN-06/78	JEN	PLAN DE EMERGENCIA EN CENTRALES NUCLEARES DE POTENCIA (EMERGENCY PLAN FOR NUCLEAR POWER PLANTS)	publicada Febr./79
GSN-07/78	JEN	CRITERIOS SOBRE LA SEGURIDAD FISICA DE LAS INSTALACIONES NUCLEARES. (INDUSTRIAL SECURITY AT NUCLEAR POWER PLANTS)	publicada Marzo/79
GSN-08/78	JEN	DOCUMENTACION PARA LA SOLICITUD DEL PERMISO DE EXPLOTACION DEFINITIVA. (APPLICATION TO DEFINITIVE OPERATION PERMIT)	publicada Nov./78
GSN-09/78	JEN	PROGRAMA DE VIGILANCIA RADIOLOGICA AMBIENTAL PARA CENTRALES NUCLEARES DE POTENCIA. (ENVIRONMENTAL MONITORING PROGRAMME)	publicada Nov./78

.../...

REF:	SPONSOR	TITLE	ESTADO
GSN-10/79	JEN	GUIA PARA LA ELABORACION DE INFORMES ANUALES DE LAS INSTALACIONES RADIACTIVAS. (GUIDE FOR THE ELABORATION OF RADIACTIVE INSTALLATIONS ANNUAL REPORTS)	Draft
GSN-11/79	JEN	VIGILANCIA DE EFLUENTES RADIACTIVOS LIQUIDOS Y GASEOSOS LIBERADOS EN CENTRALES NUCLEARES DE POTENCIA. (MONITORING OF AIRBORNE AND LIQUID RADIOACTIVE RELEASES FROM NUCLEAR POWER PLANTS)	Draft
GSN-12/79	JEN	ALMACENAMIENTO TEMPORAL DE RESIDUOS RADIACTIVOS. (TEMPORARY SOTORAGE OF RADIOACTIVE WASTES)	Draft
GSN-13/79	JEN	PROGRAMA DE PRUEBAS PRE-NUCLEARES Y NUCLEARES. (PROGRAMME OF PRE-NUCLEAR AND NUCLEAR TESTS)	Draft



TEMPORARY EVOLUTION OF THE DEVELOPMENT PROCESS OF A SAFETY GUIDE



JEN SAFETY GUIDES ELABORATION PROCESS BY NUCLEAR STANDARDS SERVICE

GENERAL SAFETY CRITERIA FOR NUCLEAR POWER PLANTS
A - SMALL - COUNTRY - APPROACH

Per B. Suhr
Inspectorate of Nuclear Installations
Roskilde, Denmark

The paper deals with the preparation of general safety rules in a small country with limited resources at a time, when a Government decision to use nuclear energy has not yet been made. It will specifically describe experiences obtained during the preparation of general safety criteria to be issued by the minister of environmental protection concerning layout, design, construction, and operation of nuclear power plants of the light water type.

Cette communication traite la préparation de règlements de sécurité généraux dans un petit pays avec des ressources limitées, à un moment où la décision gouvernementale d'utiliser l'énergie nucléaire n'a pas encore été prise.
Le rapport s'occupe surtout des expériences obtenues pendant la préparation de critères de sécurité généraux à être publiés par le Ministre de l'Environnement, concernant la disposition, le projet, la construction et l'exploitation des centrales nucléaires du type à eau légère.

1. Introduction.

The licensing procedure for siting, construction, and operation of future nuclear installations in Denmark will be regulated by the Act of May 12 1976 "Governing the safety and environmental conditions applicable to nuclear installations". However, most of this law has not and will not come into force until the Danish Parliament has passed an implementing Act following a positive decision on a Danish nuclear power programme. A positive decision on a nuclear programme will depend on a referendum, which will probably take place in 1981. Meanwhile, the authorization procedure is regulated by the Nuclear Installations Act of May 16, 1962, as amended by the Act of June 12, 1974. February the 1st, 1976 the administration of the Nuclear Installations Act (1962) was transferred from the Ministry of Education to the Ministry of the Environment. As a consequence of this change tasks of the former Atomic Energy Commission as a nuclear safety authority was assigned to the National Agency of Environmental Protection. At the same time, the Inspectorate of Nuclear Installations was transferred from the Atomic Energy Commission to this Agency.

The introduction of nuclear energy in Denmark is prepared by work on governmental nuclear safety rules on subjects as radiation, transport of nuclear materials, design, plant performance and quality assurance. Other examples are requirements to safety assessment documents and planning of licensing procedures for nuclear power plants.

2. Decision on preparation on Danish safety criteria.

The question may be raised, whether a small country like Denmark needs to develop its own general safety criteria for nuclear power plants, as it could be argued that the existing limited number of reactor suppliers essentially offer power plants in accordance with the criteria of their own country. One could therefore consider the possibility that the general safety criteria of the reactor supplier country could normally be accepted by the Danish authorities.

There are, however, differences in the general safety criteria of the different potential reactor supplier countries, and the working group which prepared the proposal for general safety criteria concerning lay-out, design, construction, and operation came to the conclusion that the nuclear power plants, which may be built in Denmark, shall fulfil requirements, which at least reflect the requirements in any of the potential reactor supplier countries. Thus it was agreed that the Danish authorities ought to work out a coherent set of criteria expressing a level of safety, which can be obtained to-day using well documented technology. As a guiding principle we found it important that the requirements for plant design and plant performance should as far as possible correspond to those of the potential reactor supplier countries.

3. Resources and sources.

This paper specifically describes the work of the Danish group, which has prepared a proposal on criteria concerning lay-out, design, construction, and operation of nuclear power plants of the light water type. The group was formed in spring 1978 of 8 members from different authorities and organisations. Also the Power companies were represented in the group. During the preparation of the criteria, the group has to a certain extent had discussions with experts outside the group. The criteria have now been forwarded to different Danish authorities and organisations for remarks.

As the task of the group has been to work out a set of general safety criteria within 15 months, it has been necessary to obtain guidance using criteria already worked out elsewhere. This is, of cource, also reasonably from other points of views. Firstly it is obvious to use - as much as possible - the work and experience, which is contained in existing criteria. Secondly it should be recalled that we have no practical experience from Danish nuclear power plants, and thirdly, as mentioned earlier, it has been a guiding principle in the preparation of the criteria that the requirements for plant performance should

as far as possible correspond to those of the potential reactor supplier countries.

Turning now to the existing criteria which have been used as background, it could be mentioned that the German criteria "Nuclear Power Plant Safety Criteria" (The German Federal Minister of the Interior, Oct. 21, 1977) and the USA/NRC criteria given in 1o CFR Part 5o has to some extent been normative. But also the Nordic recommendations for criteria "General safety criteria for the design of nuclear power plants with light-water reactors" have been used as a guide.

4. Basic Principles.

The technical content in the criteria is based on two principles, one concerning operation of the nuclear power plant (normal operation and operational incidents) and one concerning precautions against accidents.

The first one states that the lay-out, design, construction, and operation of the nuclear power plant shall be such, that plant functions are performed with a high degree of reliability and such that prescribed rules and parameter limits in Technical Specifications (e.g. pressure and temperature) are not violated in case of failures or impacts, which empirically must be assumed to occur during the lifetime of the power plant.

The other principle states that regardless of the fact that - as stated in the first basic principle - the power plant shall carry out its function with a high degree of reliability, the design of the plant shall provide for that failures may occur, or that the plant is exposed to impacts violating the rules or limits, mentioned in the first basic principle. As a consequence, the plant shall be provided with mechanical and structural safety arrangements, which shall assure against injuries in the population as a result of the types of design basic accidents and impacts specified in the criteria.

As further remark to the basic principles it could be mentioned that the stipulation of the types of failures, determining the design of the safety systems, has been done by requiring that failure types shall include those, which consist in or may lead to rupture on the primary system of the plant. These types are

a. Spontaneous rupture on the primary system.
b. Other failures with the consequence of insufficient cooling of the fuel elements.
c. Failures which imply uncontrolled heat production in the fuel (reactivity transients).

Types of failures, which shall be taken into account include therefore, but are not restricted to pipe rupture, pump failure, valve failure, failure of the control rod system, and internal missiles.

Only failure types generated within the plant have been mentioned here. There are, of course, criteria on external impacts also.

5. Examples on special efforts.

This is, of course, not the intention here to give detailed information on the content of the different safety criteria. They consist of sections as plant lay-out in general, quality assurance, requirements to process systems and safety systems, requirements to control rooms, specification of types of failures determining the design requirements to the safety systems, and conditions concerning the operation of the plant.

However, two subjects, where special efforts have been placed, will be mentioned here. The first one concerns the reliability requirements to the plant. The working group came to the conclusion that it was appropriate to state the plant reliability requirements in a separate criterium. This criterium starts with the statement that systems and components, the function of which totally or partly can be motivated from safety reasons to a

third party shall be designed, manufactured, installed, tested, and maintained in a way which assures the necessary high reliability of the functions, they shall carry out.

The criterium contains separate, general requirements to the process systems and safety systems. The stated general requirements to the process systems are that they shall be designed, manufactured, installed, tested, operated, and maintained so that operational occurrences can be controlled in a way which assures, that the automatic functions of the safety systems seldom are necessary. We found it important to stress such a characteristic of the plant.

Regarding the general reliability requirements to the safety systems, it could be mentioned that our general impression, when reading existing criteria, is that it is difficult to get a comprehensive view of the reliability requirement valid for all of the safety systems. In our terminology the safety systems are the protection system, the safety shut down system, the containment system, and the emergency core cooling system. We therefore concluded that it would be worth-while to formulate a general reliability requirement valid for all the safety systems. This requirement can then be supplemented with requirements to the individual systems, e.g. requirements concerning testing and maintenance.

The stated general requirements to the safety systems are that they shall be designed, manufactured, installed, tested, and maintained in a way which assures the necessary high reliability of the function, they shall carry out, if an accident initiating event with consequential faults occur. As provisions should among other things be used redundancy, diversity, separation, and mechanical protection against external impacts on the systems - and provisions to ensure the reliability of the different parts of the systems through quality assurance and periodic tests. These provisions shall be used to such an extent

- that the failure criteria is fulfilled

- that the system function is protected against systematic failures.

The failure criteria states that if an accident initiating event with consequential faults occur in the plant, the system must be able to carry out its functions, also in case of failure of any active or passive component in the system, or in case of an operator error (inclusive consequential faults) resulting in that one or more parts of the system looses the ability to carry out its functions.

In connection with the reliability requirements to the safety systems I should like to add, that there has been intense discussions on the problem of operator interference with the automatic actions of the safety systems, a problem which have been made topical by the TMI-accident. We have studied German and Swedish policy and rules, and in this connection the Swedish so-called 3o-minute rule could be mentioned. This rule states that all safety precautions necessary within 3o minutes after an accident initiating event has taken place shall be automatized.

The group concluded with the following criterium:

All functions of the safety systems necessary within a specified period after an accident initiating event has occured shall be automatized. The automatic functions of the safety systems must not be weakened or prevented by manual operation.

The other example I would like to mention concerns the requirements to the reactor containment. In the proposal worked out by the group the content in the criterium is of a "classical" character and relates to the well known design basis accidents.

However, great efforts are made in different countries in order to reduce the consequences of hypothetical accidents beyond the design basis accidents assumed to-day, i.e. accidents with partial or total failure of the safety systems resulting in core melt down.

The efforts have to a great extent been in the direction of considerations toward remote siting and emergency measures. However, the problem also contains the question about what can be done in order to improve the safety of the plant by reducing the consequences of core melt down accidents. Different possibilities are discussed in different countries, for example vent of the containment. However, problems arise here, because the probability for core melt down accidents with serious consequences is already very small with the design of to-day's power plants. And the question arises, how small should the probability be in order to be acceptable.

The question has been discussed intensively in the group and discussed with experts outside the group, and it has considered whether it would be possible to circumvent the problem and establish the requirement that the containment in case of failure of the mechanical safety systems shall be able to reduce the release of radioactive materials in the surroundings to an extent that

- the population in the near-by surroundings will not be exposed to acute health effects.

- the number of illnesses and genetic defects will not increase significant in the surrounding population taken as a whole.

We are not able to solve the problem to-day, but we find, that this is one of the questions, which has to be solved in the years to come.

Second Period of Discussions on Session III

F.J. Turvey (Ireland) (To J.L. Butragueño)
You say that the present nuclear standards have been produced because of an identified need by the JEN. Can you explain exactly what this need is.

J.L. Butragueño (Spain)
We consider it is useless to write guides on those areas in which it is necessary to have a good engineering experience, as it is, for example, in reactor design. We cover this area with standards and guides from the main supplier country. But there are other areas in which both the developing of the general regulations established and our own experience make it necessary or appropriate to write national guides and stan dards. These areas are, for example, qualifications of candidates to be an operator and supervisor, radiological surveillance or information to be provided on applying for an operation permit.

H. Matulla (Austria) (To. P.B. Suhr)
Do you think that if there will be enough data on reliability of components in the future, a licensing body will accept nuclear power plant safety systems on a pure probabilistic basis alone?

P.B. Suhr (Denmark)
If there is enough data, and the vendor can prove that the reliability of the data is sufficiently good, yes, then it is possible to license the plant safety systems on a pure probabilistic basis. The problem is that up til now, no one has been able to license safety systems on a pure probabilistic basis because of lack of data. The probabilistic method is a method of the future.

F. Cogné (France) (To. M. Perelló)
Me référant à votre première conclusion, où vous dites que de bons standards réduiront considérablement le temps du licensing review, je voudrais vous demander si vous entendez par cette phrase que l'essentiel du licensing, dans l'avenir, consistera en l'examen de conformité des installations à ces standards. Si tel est le sens de cette phrase, je crains que ceci ne conduise directement qu'à des accidents de type TMI. Pour ma part je pense que les standards et guides de sûreté ne doivent pas dispenser d'effectuer un technical assessment complet; ils sont plus utiles pour les constructeurs que pour les autorités de licensing.

M. Perelló (Spain)

J'aimerais ajouter un commentaire à la question posée par M. Cogné au sujet du fait qu'il vaut mieux faire une évaluation lorsque l'on analyse les problèmes plûtot que de se borner à appliquer ce que disent les guides. Je suis d'accord sur ce point, mais estime que c'est toujours un moindre mal d'adopter une solution intermédiaire entre les trois options fréquentes d'évaluation en matière de sûreté nucléaire, à savoir: la méthode de normalisation, la méthode d'installation de référence, la méthode d'analyse.

L'idéal serait évidemment de faire l'évaluation en tenant compte des trois solutions.

J.S. MacLeod (UK) (To P.B. Suhr)

In Mr. Suhr's paper the criteria is stated that operation intervention after an accident event has occurred shall be automated. Does this mean that any intervention, even if seen to be favourable, is not possible?

P.B. Suhr (Denmark)

In the criterium is stated that if the safety systems have been activated after an accident initiating even, the functions of the safety systems must not be weakened or prevented by manual operation. Of course it should always be possible for an operator to activate any of the safety systems if he feels that it is favourable.