Proceedings of the 2nd CSNI Specialist Meeting on Simulators and Plant Analysers

Espoo, Finland, 29 September – 2 October, 1997

Edited by
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Organized by
OECD Nuclear Energy Agency

in collaboration with
VTT

 TECHNICAL RESEARCH CENTRE OF FINLAND
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Preface

The Second CSNI Specialist Meeting on Simulators and Plant Analyzers: Current Issues in Nuclear Power Plant Simulation was held in Hanasaari, Espoo, Finland, from September 29 through October 2, 1997. It was organised by OECD Atomic Energy Agency, CSNI Principal Working Group on Coolant System Behaviour (PWG2), Task Group on Thermal Hydraulic Applications (TG-THA), in co-operation with Technical Research Centre of Finland supported by Ministry of Trade and Industry and sponsored by IVO Power Engineering Ltd. and Teollisuuden Voima Ltd.

The meeting attracted some 90 participants from 17 countries. A total of 49 invited papers were presented in the meeting in addition to 7 simulator system demonstrations. Ample time was reserved for the presentations and informal discussions during the four meeting days. The previous meeting held in Lappeenranta, Finland in 1992 collected some 85 participants from 12 countries, presenting a total of 40 papers.

The meeting was structured into 6 sessions covering the important aspects of development and use of simulators and plant analyzers:

Session I: New objectives, requirements and concepts
Session II: Trends in simulation technology
Session III: Training and human factor studies using simulators
Session IV: Modelling techniques
Session V: Plant analysis applications
Session VI: Simulator validation and qualification

In addition to formal lecture sessions, a show room was provided with demonstrations on simulators and simulator development tools.

As the chairman of the Program Committee I want to thank all the individuals and organisations that have contributed to the successful and interesting meeting.

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Program Committee

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Committee on the Safety of Nuclear Installations

The NEA Committee on the Safety of Nuclear Installations (CSNI) is an international committee made up of scientists and engineers. It was set up in 1973 to develop and coordinate the activities of the Nuclear Energy Agency concerning the technical aspects of the design, construction and operation of nuclear installations insofar as they affect the safety of such installations. The Committee's purpose is to foster international cooperation in nuclear safety amongst the OECD Member countries.

CSNI constitutes a forum for the exchange of technical information and for collaboration between organisations which can contribute, from their respective backgrounds in research, development, engineering or regulation, to these activities and to the definition of its programme of work. It also reviews the state of knowledge on selected topics of nuclear safety technology and safety assessment, including operating experience. It initiates and conducts programmes identified by these reviews and assessments in order to overcome discrepancies, develop improvements and reach international consensus in different projects and International Standard Problems, and assists in the feedback of the results to participating organisations. Full use is also made of traditional methods of co-operation, such as information exchanges, establishment of working groups and organisation of conferences and specialist meetings.

The greater part of CSNI's current programme of work is concerned with safety technology of water reactors. The principal areas covered are operating experience and the human factor, reactor coolant system behaviour, various aspects of reactor component integrity, the phenomenology of radioactive releases in reactor accidents and their confinement, containment performance, risk assessment and severe accidents. The Committee also studies the safety of the fuel cycle, conducts periodic surveys of reactor safety research programmes and operates an international mechanism for exchanging reports on nuclear power plant incidents.

In implementing its programme, CSNI establishes co-operative mechanisms with NEA's Committee on Nuclear Regulatory Activities (CNRA), responsible for the activities of the Agency concerning the regulation, licensing and inspection of nuclear installations with regard to safety. It also co-operates with NEA's Committee on Radiation Protection and Public Health and NEA's Radioactive Waste Management Committee on matters of common interest.
Organisation for Economic Co-operation and Development

Pursuant to Article I of the Convention signed in Paris on 14th December 1960, and which came into force on 30th September 1961, the Organisation for Economic Co-operation and Development (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; and
- to contribute to the expansion of world trade on a multilateral, non-discriminatory basis in accordance with international obligations.

The original Member countries of the OECD are Austria, Belgium, Canada, Denmark, France, Germany, Greece, Iceland, Ireland, Italy, Luxembourg, the Netherlands, Norway, Portugal, Spain, Sweden, Switzerland, Turkey, the United Kingdom and the United States. The following countries became Members subsequently through accession at the dates indicated hereafter: Japan (28th April 1964), Finland (28th January 1969), Australia (7th June 1971), New Zealand (29th May 1973), Mexico (18th May 1994), the Czech Republic (21st December 1995), Hungary (7th May 1996), Poland (22nd November 1996) and the Republic of Korea (12th December 1996). The Commission of the European Communities takes part in the work of the OECD (Article 13 of the OECD Convention).

NUCLEAR ENERGY AGENCY

The OECD Nuclear Energy Agency (NEA) was established on 1st February 1958 under the name of the OEEC European Nuclear Energy Agency. It received its present designation on 20th April 1972, when Japan became its first non-European full Member. NEA membership today consists of all OECD Member countries except New Zealand and Poland. The Commission of the European Communities takes part in the work of the Agency.

The primary objective of the NEA is to promote co-operation among the governments of its participating countries in furthering the development of nuclear power as a safe, environmentally acceptable and economic energy source.

This is achieved by:

- encouraging harmonization of national regulatory policies and practices, with particular reference to the safety of nuclear installations, protection of man against ionising radiation and preservation of the environment, radioactive waste management, and nuclear third party liability and insurance;
- assessing the contribution of nuclear power to the overall energy supply by keeping under review the technical and economic aspects of nuclear power growth and forecasting demand and supply for the different phases of the nuclear fuel cycle;
- developing exchanges of scientific and technical information particularly through participation in common services;
- setting up international research and development programmes and joint undertakings.

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

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Meeting Summary

1. Sessions

The meeting was structured into six lecture sessions followed by a session devoted to discussion and exchange of opinions. The lectures covered the important aspects of development and use of simulators and plant analyzers:

Session I: New objectives, requirements and concepts
This session covered the progress experienced since the 1st simulator meeting and tried to address the changing role of simulators based on the changes in users' needs and developing possibilities.

Session II: Trends in simulation technology
This session was reserved for studying the current trends in the simulation technology: software environments, visualisation, simulator configuration tools, programming languages and computer systems.

Session III: Training and human factor studies using simulators
This session was created for studying the status of different uses of simulators such as educational simulators, human factor studies and integrated safety assessment in addition to traditional training. Regarding to the severe accidents, a question was raised whether the simulator use should be for training or education.

Session IV: Modelling techniques
The session on modelling techniques was included to cover recent developments in the modelling techniques applicable to training simulators and plant analyzers. One of the main interests were the interconnection of analysis level models in real time applications.

Session V: Plant analysis applications
This session was planned to cover the various applications of plant analyzers in plant design evaluation, system qualification, development of operation procedures and safety analysis. The new concepts such as multifunctional simulators and Living Plant Analyzers supporting many of the above mentioned applications and adding features like evaluation and testing of new control systems and operator support tools.

Session VI: Simulator validation and qualification
This session concentrated on simulator validation and qualification procedures followed in the training centers as well as on the problems encountered in areas such as severe accidents and accident management. The requalification of the simulator was also a topic of discussion.
2. Paper summaries

Session I  New objectives, requirements and concepts

A total of four papers were presented in this session. The first paper 'Summary of the 1st Specialist Meeting on Simulators and Plant Analyzers held in Lappeenranta' (VTT) covered the results of the previous meeting in 1992 and compared the visions presented in the meeting with the developments actually realised. Many of the simulator concepts and tools presented in the previous meeting still exist in a more mature form. Specifically the gap between the analysis codes and the simulator models is disappearing, much due to the dramatic increase of affordable computer power available today. The developments also include successful interconnection of thermal hydraulic and reactor kinetics models as well as the appearance of multifunctional simulators and research on human factors, all foreseen in the previous meeting as part of the desirable co-operation between specialists of different disciplines.

The second paper 'Human Machine Interaction Research Experience and Perspectives as Seen from the OECD Halden Reactor Project' (Halden) covered the research experience of Halden Reactor Project. The major topics were the studies and related methodological development on the operator cognition and information processing performance in control room environment (HAMMLAB), the performance and problems related to information presentation and the development of plant surveillance and support systems. A Virtual Reality Center has been founded as a complementary extension of HAMMLAB for studies of existing and future control rooms. As a response to increased interest in human factors research, a project has been launched to create a new facility HAMMLAB-2000, equipped with realistic PWR and BWR simulators.

The paper 'Regulatory Perspectives on NPP Simulator Applications' (SKI) considered the regulatory perspectives on the NPP simulator applications. Traditionally the focus of interest of the regulatory bodies has lain on three major topics: the operator training, the safety analyses using advanced plant analyzers and the procedures for preparedness and emergency operating procedures. In the future, the major issues will be the large NPP modernisation and back fitting projects being carried in many countries and their relation to the public conscience of the risks involved. The confidence among public erodes quickly, if problems arise causing unplanned interruptions or shutdown due to safety problems. To prevent this, the measures taken must be explainable: robust and simple. In case of transients, incidents or anomalous plant behaviour, a careful analysis using advanced plant analyzers or other modern analytical tools is considered essential. Answers should also be found to 'What if'-types of questions, trying to find the often small, trivial looking incidents that have the tendency to develop into accidents.

The last paper 'Operator Aids for Severe Accident Management' (AVN) was a summary of the second OECD specialist meeting on Operator Aids for Severe Accident
Management (SAMOA-2) held in Lyon, 8-10 September, 1997. The scope of the meeting covered operator aids for accident management, analysis methods and relevant simulation tools for operator training. The general conclusions indicate that the development and implementation of operator aids for accident management is in progress but proceed slower than expected. The training for severe accident management (SAM) is gaining acceptance, while there is still a debate on the orientation of the training: skill or knowledge oriented. The tools for SAM training are still in infancy. For concrete results, international collaboration should be increased.

**Session II  Trends in simulation technology**

The session 'Trends in Simulation Technology' dealt with aspects of new technologies in the development of simulators and NPAs. The first three papers, 'ALICES: an Advanced Object-Oriented Software Workshop for Simulators' (CORYS T.E.S.S.), 'CISO: Charter of Integration for Simulation Openness' (EDF/SEPTEN) and 'APROS - a Multifunctional Modelling Environment' (VTT) were concerned with the modelling environment. The efforts are directed towards providing homogeneous modelling environments, establishing model interfaces in an object-oriented fashion, and enabling reuse of components, thus reducing simulator development time and costs. In particular, the requirements of developers concerning user interfaces which provide high flexibility in adapting existing system designs and models are addressed. A step further in the direction of a multifunctional environment for a wide range of various applications including non-nuclear simulators is taken by APROS. Its layered architecture allows to model increasingly complex systems by using the features of lower layers.

In paper 'GRASS - the Graphic Simulation System' (KFKI) a slightly different approach of constructing source code directly from a user defined graphic network was described, with reusable graphic components instead of predefined models. The system is integrated with the real-time executive and the data base of the simulator.

In paper 'The Trend towards Windows NT for Use in Simulation' (RNI) a strong case was made for utilising PC's running Windows NT in simulators. The main advantages are seen in the availability of well-proven third party tools at costs much lower than in the UNIX world, decreased operating and maintenance costs, as well as the ready access to spare parts. The increasing number of simulators in operation or ordered on this basis is proof to the strong impact of Windows NT in the simulator world.

A study in paper 'Using Intranet with Simulation' (GSE) on network-wide simulation as to the possibilities offered by the Internet and the Internet-aware programming language Java, along with the prototype of a distributed client-server architecture, showed the feasibility of such an approach.

The paper 'Enhanced Productivity of Simulation Engineers' (GSE) made clear, how productivity of simulation engineers has steadily increased in accordance with advancing technology. It is claimed that with the modern modelling environments and
tools available today, simulation engineers may concentrate on their proper tasks, without the burden of computer science heavy on their shoulders.

In summary, the impact of the object-oriented approach in software development has reached simulator development. Strong efforts are seen to provide standard interfaces to reuse models written in different styles and languages, to group objects to form more complex systems, and to provide user interfaces relieving the simulation engineer of many software engineering problems. According to the general trend in migrating applications to Windows NT, the first simulators are appearing on a PC-basis. For the future, the direction of client-server architectures, which has already been realised in some plant analyzer applications, will be taken further to distribute simulations on the Internet. With respect to openness of simulator environments and standardised model interfaces, it remains to be seen if these concepts will extend beyond the scope of single simulator manufacturers.

Session III  Training and human factor studies using simulators

A total of 11 papers have been presented, covering the state of the art in training simulators and human factor issues. According to the main focus of these presentations, they can be grouped in the three following categories:

7 papers 'Training of Nuclear Power Plant Personnel in the German Simulation Centre' (GSC), 'Nuclear Plant Analyzer: An Efficient Tool for Training and Operational Analyses' (Tractabel), 'Multifunctional Optimised Scope Simulators in Central and Eastern Europe' (CORYS T.E.S.S.), 'Leningrad NPP Full Scope Simulator - New generation Tool for Training and Analysis' (Kurchatov), 'Leningrad NPP Full Scope and Analytical Simulators as Tools for MMI Improvement and Operator Support Systems Development and Testing' (Kurchatov) and 'An Interactive Graphic Simulator (MAAP4) for Garoña Nuclear Power Plant: Development and Applications' (Univ. Cantabria) are related to the progress and state of the art in simulators used for training purposes,

2 papers 'An Operator Self-training System Based upon the Emulation of Instructor Skill (Ansaldo Nuclear) and Tree Simulation Techniques for Integrated Safety Assessment' (Consejo de Seguridad Nuclear) where the use of simulators has been proposed in connection with the development or the verification of plant operating/emergency procedures and

2 papers 'HAMMLAB 2000 for Human Factor’s Studies' (Halden Reactor Project) and 'Development of a Research Simulator for the Study of Human Factors and Experiments' (TEPCO) dealing with the use of simulators for human factor experiments.

From the first group of presentations appears that quite a substantial progress has been made in the last few years in the performance, accuracy and capability of simulators in the training area. Another interesting aspect is that also accident condition are currently addressed in training programs and that consequently training simulators are now
capable to cope with abnormal plant conditions, at least up to Design Basis Accidents. The next challenge is to evaluate the possibility (and the convenience) to have training simulators able to go into severe accidents conditions. Although some research has been done in that direction (an example is the Leningrad training simulator, from Kurchatov Institute), it seems not be agreement on the opportunity to include severe accident conditions in operator training programs.

A topic, which has regrettably been left completely out is the simulator capability for training in shutdown conditions. Although in recent years, the risk of accidents during outages has been evaluated to be comparable to the risk at power conditions (with associated increased human error probabilities), apparently no special efforts have been made in extending simulator capabilities to cope with the special conditions which exist in shutdown. An exception is the full-scope simulator at Doel, where Tractebel has developed a model for mid-loop operation (see Session IV: The Implementation of a Mid-loop Model for Doel 1 and 2 Training Simulator). It is suggested that this issue will be addressed in the near future.

The use of simulators in connection with procedure development/verification is a relatively new issue and, from the presentations on this topic, it seems quite promising, especially when the target is the verification of Severe Accidents Management Guidelines (SAMG). However, this leads back to the need of simulators able to dig well into severe accidents.

Finally, we had a couple of presentations related to the use of simulators for human factors studies. Here the possibility and convenience to run human factor experiments in simulated control rooms has been presented. It includes the evaluation of operator performance in different simulated environments and abnormal conditions. We expect to see in the near future a number of interesting experiments evaluating the usefulness of the many operator support systems which are in development and it is not clear now if and at what extent they could improve the operator understanding during abnormal plant conditions.

**Session IV  Modelling Techniques**

This session offered an outlook on the coupling of qualified neutronics codes with system codes. There is a general agreement that for many transients, it is necessary to use a 3D neutron kinetics model coupled to a thermal hydraulic model in order to obtain satisfactory results. This need coincides with the fact that one may take advantage of the increase in computing power that has become available.

Five papers 'Coupling of 3D Neutronics Models with the System Code ATHLET' (GRS), 'Interfacing High-Fidelity Core Neutronics Models to Whole Plant Models' (Nuclear Electric), 'Neutronic Aspects of the THOR Core Dynamic Model for Training Simulators' (Scandpower), 'Coupled Neutronics and Thermal Hydraulics Modelling in Reactor Dynamics Codes TRAB-3D and HEXTRAN' (VTT) and 'APROS 3-D Core
Models for Simulators and Plant Analyzers' (VTT) addressed this concern and presented the ongoing activity in Germany United Kingdom, Norway and Finland.

Three papers clearly indicated the current trend to extend the scope of the simulators. The first paper 'Effective Modeling of Hydrogen Mixing and Catalytic Recombination in Containment Atmosphere with an Eulerian Containment Code' (Ansaldo) addressed the modelling of Hydrogen Mixing and catalytic recombination in containment atmosphere. Coupling an Eulerian code with a CFD code has been considered in order to predict effectively (with reasonable computer running time) and accurately (i.e. with a high degree of confidence) the natural convection flow patterns and the distribution of steam and gases in the containment.

The second paper 'The Implementation of a Mid-loop Model for Doel 1 and 2 Training Simulator' (Tractebel) presented the adaptation of an existing full scope simulator in order to simulate some mid-loop operation transients.

The third one 'Latest Improvements on TRACPWR Six-Equations Thermohydraulic Code' (Technatom) presented the latest improvements on TRACPWR six equations code that has been adapted in Spain for implementation in simulators. Beside the code speed up and code platform downsizing the scope has been enhanced for mid-loop operation and for modelling VVER and PHWR.

The paper 'The IMPACT Super-Simulation Project for Exploring NPP Fundamental Phenomena' (NUPEC) presented this ambitious Japanese project (a ten-years program). IMPACT is the name of a program which will perform full-scope, detailed calculations of physical and chemical phenomena in a nuclear power plant for a wide range of scenarios. The main modules are, the Human Interface, the Analysis System, the Data Base, the Knowledge Base and the control system that supervises the whole system.

The paper 'Integration of ANTHEM Thermal Hydraulic Model in ROSETM Environment' (CAE) presented the project to integrate a two-phase thermal hydraulic model into a Real-time Object-oriented Simulation Environment (ROSE).

The paper 'Evaluation of Two-Fluid and Drift Flux Thermohydraulics in APROS Code Environment' (VTT) presented an evolution of two-fluid and drift flux model capabilities.

The paper 'A Nodalization Study of Steam Separator in Real Time Simulation' (GSE) illustrated the nodalization effect on the results for some typical BWR transients.

**Session V  Plant Analysis Applications**

In addition to the well-known use of simulator for training purposes, this session was devoted to a lot of questions concerning the various parts of the NPP operation which may be solved on simulators or plant analyzers rather than on real plants: design evaluation, procedure validation, system testing, accident management. One must,
however, keep in mind that such applications can be possible only if the simulator meets the appropriate requirements concerning, in particular, its simulation range (adaptability to plant specific features) and its pertinence.

The paper 'Simulate-3 Core Model for Nuclear Reactor Training Simulators' (GSE) presented the work realised on the main 3D neutronic module SIMULATE-3K, the Studvisk core model. The modifications concern real-time computation, using parallelization techniques (POSIX threads of UNIX or Windows NT), and lead to a new module SIMULATE-3R, utilisable for training. The adaptations have been made for BWR and PWR Core Management System (CMS).

The paper 'ATLAS: Applications Experiences and Further Developments' (GRS) presented what has already been done on the GRS-developed ATLAS plant analyser: realisation of supplementary modules like Reliability Advisory System, Procedure Analysis and diagnostic system for SGTR accidents and implementation (in complement to ATHLET thermohydraulics code) of RAOC confinement behaviour and MELCOR severe accident codes. This allows a very wide range of applications of such a plant analyzer: the adaptation to the plant specific features has already been done for 4 plants (2 PWRs and 2 BWRs) and is under way for 3 others (1 PWR, 1 BWR and 1 Russian VVER 1000/230). In addition some developments, like implementation of Windows NT, tracking simulator and design of a multimedia Analysis Center are planned.

The paper 'SAPHIR: a Simulator for Engineering and Training on N4-type Nuclear Power Plants' (Framatome) described the new simulator SAPHIR, developed by Framatome for the French N4 reactor type. Advanced codes, like TRACAS for the primary circuit and GVAXIAL for the secondary side of the steam generator, are used for the modelling of the main systems and simulation workshop, for the representation of the fluid, electrical instrument and control networks; the man-machine interface has been also substantially improved.

The paper 'The RELAP5-Based NPA of the VVER (440/213) Type Paks NPP' (KFKI), presented the work realised performed through a co-operation between Tractebel and KFKI-AEKI, in order to build the NPA interactive graphical tool, using the RELAP5 code, for improving the knowledge of system behaviour and alarms safeguard systems response during transients like scram and loss of primary coolant.

The paper 'NPA Applications' Development in the Nuclear Safety Authority Framework' (SNSA) presented what has been done, in collaboration with Tractebel, for realising the first version of a plant analyzer for the Krsko NPP. The RELAP5 (version Mod 3.2) code was used for achieving in 1996 a tool used now for training and for improving the knowledge about the classical design accidents. The coupling with the severe accident analysis code MELCOR is foreseen in order to improve substantially the simulation range.

The paper 'CATHARE Approach Recommended by EDF/SEPTE for Training (or other) Simulators' (EDF) presented the several steps of simulator implementation in
EDF: at first (1980 - 85 design) full-scale simulators, based on DEFI-2 thermal hydraulic codes, about ten years later SIPA (and the SIPACT's) and Fessenheim and Bugey full scale simulators, based on the CATHARE-Simu and CATHARE 1 thermal hydraulic codes, allowing in particular to simulate large breaks. Even improved simulators, based on the fine-modelling of CATHARE 2, are now planned for the next years: the 10 million dollars project SCAR will allow to have, on full-scale simulators, the possibility of processing all operating conditions (except for severe accidents, leading to core degradation like melting).

The paper 'Use of Simulators for Validation of Advanced Plant Monitoring Systems' (Tractebel) described how the Doel simulator was used for the validation of a process monitoring and supervision systems, DIMOS. Through an extensive testing campaign of two versions (one alarm-masking and one non alarm-masking) of this system, the importance of the alarm treatment for the operators was put on evidence.

In the paper 'CAMS as a Tool for Identifying and Predicting Abnormal Plant State Using Real-Time Simulation' (Halden Reactor Project) it was described how the modular CAMS systems, in particular the tracking simulator module, can provide assistance for the assessment of the future development of accidental situations and in planning mitigation strategies. We may note that, even if this project started at Halden only in 1993, the realised prototype was successfully used during a Swedish crisis drill already in 1995.

The paper 'An Intelligent Diagnostic Aid (IDA) Based upon the Simulated and Operational Experience' (Ansaldo) presented an intelligent system providing help for accidental situation diagnostics, combining expert system techniques, for issuing and managing deduction rules with a NPP simulation tool, adapted from the LEGO code. This system was developed for the Sampoierdarena mixed gas-electrical power plant and is now to be adapted for a VVER 1000.

In the paper 'Simulation Strategy for Managing Nuclear Emergencies at Jose Cabrera NPP' (Union Fenosa Ingeniera) it was shown how the simulation strategy helped in defining management orientations for the Jose Cabrera NPP, especially for the case of nuclear emergencies, and how simulation experience can boost the development of 'general purpose tools'.

The paper 'Experiences of APROS in Nuclear Power Plant Safety Analyses' (IVO) showed, how the APROS simulator took a major part in the revision of the Loviisa plant safety analysis, after the decision of power uprating program (up to 1500 MW as nominal thermal power) implementation; APROS was used for the simulation of the classical accidents (LOCA, ATWS, SGTR,...) with quite satisfactory results. The utilisation experience shows that the fine 6-equation model is only necessary for a small number of transients (such as LBLOCA); for the other operational transients a 5-equation model is sufficient.

It appears clearly, in conclusion, that the 'common denominator' for simulator utilisation is still operator and/or expert training; however some countries are now using such tools
for more advanced purposes, like accident management and/or probabilistic risk analysis.

In that perspective, the need of simulator possibility improvement for achieving these goals successfully has been clearly identified; in general, safety analysis requires much more pertinence and simulation range than (pre-programmed) training. Finally, the use of some expert system techniques seems of quite great interest for a highly efficient crisis management system.

**Session VI  Simulator Validation and Qualification**

The first paper in the session '3D Core Model for Simulation of Nuclear Power Plants: Simulation Requirements, Model Features, and Validation' (Thomson) described the requirements, features and validation of a 3D core model for four nuclear power plants with Siemens reactor control and protection systems, which consist of three staggered systems acting on the control rods imposing very stringent requirements on the core and primary coolant system simulation models. The 3D core model development concentrated on achieving the best possible match between simulation model and the design codes applied in fuel management and safety analyses. The resulting core model has ~3700 3D core cells. The simplified thermal hydraulic calculation is also carried out for each cell. The validation included stand-alone 'separate effects' and global tests as well as coupled global tests in the complete simulator environment.

The paper 'The Use of SIPA 2 Simulator for Safety Studies: Experience Feedbacks and Future Developments' (IPSN) described the use of SIPA 2 simulator for expert training, development of accident management aids and safety studies during the last 4 years. The extensive use resulted in many proposals for simulator model development and qualification, applicable to many similar systems such as multifunctional simulators.

The paper 'A Verification and Validation Program for Simulators for Soviet-Designed Nuclear Reactors' (BNL) describes the verification and validation program for 11 simulators (both full-scope and analytical) for Soviet-designed nuclear reactors in Russia and Ukraine. The program includes V&V course for the NPP staff and the description of a acceptance test procedure.

The last paper 'Interactive Graphical Analyzer Based on RELAPS/Mod 3.2 NPA' (P.M.S.A.) describes an interactive graphical analyzer based on Relap5/Mod 3.2 -NPA. The model describes the R5 model of a PWR, including primary coolant system, steam generators, secondary system up to turbine and condenser, and the feed water system. The relevant protection and control systems are included. Some modifications have been made in the graphics interface in order to improve the analysis and teaching functions of the NPA. The validation process is underway.

The user experience was in general positive and training using simulators is now given to new categories of personnel (safety authorities, crisis managers, safety analysts). Verification of simulators by assuring the equality of plant and simulator environments
important and fairly easy to achieve. Validation of models may be done by comparisons with advanced computer models (transients, accidents) and plant data. It is important that reactor physics data are validated against true plant behaviour. Emphasis is also laid on comparisons with plant transients.

**Session VII  Discussion**

The Chairman summarised the major points of the meeting regarding:

- need for maintaining expertise (analysis, modelling, what about available funds)
- training and education (who, how and to what extent include severe accidents?)
- control rooms and operator support tools (human factors, operator performance)
- simulators and safety analysis tools: all-in-one -tools, qualification (simulator benchmarks?)
- future activities: develop current technology or look for new approaches?

The following comments were made and issues raised during the discussion:

A question was raised if we do too much of modelling? There may be some duplication with work of stand alone code developers. Although it may lead to diversity of approaches, too much diversity may become counterproductive. There is a related problem regarding uncertainty. The importance and awareness of it should be increased because too often we are taking the simulator results as “true” plant response, which may not be the case.

There is a trend of merging various codes and/or subroutines that were validated separately. It is also necessary to validate the 'combined' codes, since putting together various modules require some changes that may effect the validation of the final product. There is still a difference in an engineering "culture" and practices between the TH and neutronic code developers. More contact and joint meetings are needed to bridge this gap.

There was a suggestion to establish a possible benchmark for simulators. It is not clear how such a benchmark could be defined, however an example was given of an AI benchmark established about 2 years ago. It was agreed, however, that such a benchmark would be very useful in particular given ongoing MSLB benchmark on TH/neutronic coupling. KOLA simulators was mentioned as a possible model for such an exercise. A recommendation was made to organise a one day brain-storming session to discuss possibility of a simulator benchmark. Such a meeting should include representatives from Halden, PWG4 and IAEA.

An observation was made that the simulator applications have changed during the last 5 years, i.e., there is more training, safety analysis and support for regulators. In addition, the simulators are changing faster and faster leading to a possibility of applications in
the areas of MMI and human factors. Question were raised regarding possible establishment of standards for GUIs.

Difference in opinions existed regarding whether severe accidents should be included in simulators' models, in particular in full scope simulators. The benefits of having SA capabilities for operator training were questioned. Most of participants agreed that, although the emphasis should be on plant 'normal' operations and transits, at least some aspects of SA should be included. The SA training should be aimed not so much to operators, but rather to others like technical support center personnel, inspectors and regulators.

There is still an issue about usefulness of the real-time simulators in the control room during accident conditions (but not in normal operations), i.e., the operators may not be able, or would not have the time to take advantage of the simulators results.

An interesting suggestion was made to develop a simulator for plant maintenance procedures practices.

The themes of SAMOA meetings appear almost as a subset of this series of meetings. In the future, these meetings could well be combined. This observation and recommendation was expressed several times.

3. Major conclusions of the meeting

The major conclusions of the meeting can be summarised as follows:

- The role of simulators is changing and the applications are becoming more diverse.
- The differences between training simulator and plant analyzer software are disappearing.
- It would be useful to establish a basis, or a set of rules for comparison of simulators. A small group of experts should be formed to study feasibility and benefits of such an exercise and possible comparison criteria.
- The themes of SAMOA meetings appear almost as a subset of this series of meetings. In the future, these meetings could well be combined.
- A suggestion was made to develop a simulator for plant maintenance procedure practices.
Welcome remarks
Andrzej Drozd
OECD/NEA

Ladies and Gentlemen.

On behalf of Director General of the OECD Nuclear Energy Agency, Mr. Luis Echavarri, I would like to welcome you to the 2nd OECD Specialist Meeting on Simulators and Plant Analyzers. Also, I would like to thank Ministry of Trade and Industry of Finland for the sponsorship and our host, the Technical Research Centre of Finland (VTT) and its staff, for their great help in organising the meeting.

This Specialist Meeting (SM) is a part of the activities of the Principal Working Group 2 (PWG2) on Coolant System Behavior in the area of thermal-hydraulic applications. The First SM on simulators took place in June 1992, also hosted by VTT, indicated a variety of important applications of simulators and plant analyzers. Other related meetings include the OECD/CSNI Workshop on Transient Thermal-hydraulic and Neutronic Codes Requirements, November 1996, in Annapolis, and the OECD/CSNI Specialist Meeting on Advanced Instrumentation and Measurements Techniques, March 1997, in Santa Barbara.

One of the simulators related activities is a development of the second generation of thermal-hydraulic codes coupled with neutronic models. As discussed at the Annapolis meeting, there is a need for a two-phase, two- and three-dimensional hydrodynamics code based on existing computational fluid dynamics (CFD) techniques. Such an advanced code needs to be coupled with the three-dimensional neutronics consistent with the level of detail of thermal hydraulic models. There is a number of activities in this area in various countries. The one activity I would like to mention is the PWR-MSLB benchmark, based on TMI-1 data, currently being performed within the PWG2 in co-operation with the Nuclear Science Committee.

Another issue discussed in Annapolis, directly related to the topic of this meeting, is a need for "user friendly" codes, and in particular the need to develop a graphical user interface (GUI) compatible with the users needs. Again, there are various GUI models under development in several CSNI member countries. An important work in this area is also carried out under the OECD sponsored HAMMLAB project in Halden.

Finally, there is a long term issue of codes verification and, consequently, software validation procedures. As we all know, even the most efficient numerical techniques and the fastest computers are "useless" if the physical models employed in the codes are inaccurate. This very important issue is being continuously discussed within the PWG2. The CSNI established a thermal hydraulic data bank in support of the Integral Test Facilities Validation Matrix. The data bank is currently being expanded to include
the Separate Effect Tests Validation Matrix. There is also a discussion within the PWG2 to establish an OECD Project in support of thermal hydraulic experimental needs.

I am sure that all of these topics will be discussed here, in Espoo. At this point I would like to thank again our Finnish hosts, and in particular Dr. Mattila, Mr. Tiihonen and their colleagues, for an excellent preparation of this meeting. I wish all of you many interesting and fruitful discussions.
Opening remarks

Lasse Mattila
Research Manager, Nuclear Energy
VTT Energy

It is a great pleasure to me on behalf of the Technical Research Centre of Finland, VTT, to welcome you to Finland and Espoo. VTT’s main campus in Otaniemi is only some three kilometres from here. In Otaniemi, VTT works in close co-operation with the Helsinki University of Technology, the largest of its kind in Finland. There are also several specialised research institutes and high-technology companies located in Otaniemi area.

VTT employs some 2700 people, almost 60% having a university degree. There has never been any dedicated nuclear research centre in Finland. However, nuclear energy research is an important part of VTT’s activities and is closely coupled with other applications. We operate nowadays as a concern of nine quite independent research institutes. Indeed, six out of these nine institutes are now regularly involved in nuclear R&D or provide services for the nuclear industry in Finland and abroad.

In several areas, nuclear applications have been the driving force when VTT has developed new capabilities, with important spin-off effects. Simulation of dynamic industrial process systems is certainly one of the best examples in this respect, Fig. 1. In early 1970’s, immediately after nuclear power plants were ordered in Finland, we started to carry out accident analyses, with the first versions of the RELAP code, for example. Analogue simulators were built already in the late sixties, followed by hybrid simulators in seventies. We also had an important role in the building of the full-scope training simulator commissioned in 1980 for the Loviisa NPP. In 1980’s we started to look at the simulation of combustion power plants and in recent years we have had strong interest in paper mills, both areas being very important for the Finnish industry.

We who are interested in the simulation of nuclear power plants can also benefit of spin-in from other areas of technology. The huge progress in the computational fluid dynamics, CFD, helps us to improve our thermal-hydraulic modelling, and latest visualisation and virtual reality techniques offer possibilities to enlarged scope of simulation in training, in maintenance for example. Also, simulators are very useful as test beds when we have to modernise the I&C systems of nuclear power plants by introducing programmable automation systems and new control room features.

About ten years ago we draw the first version of a picture illustrating the different numerical simulation tools available for nuclear power plant design, safety analysis and
training. Fig. 2. At that time we had quite distinct categories of sophisticated accident analysis codes, fast-running simplified models for simulators and some early attempts to combine fidelity and comprehensiveness of the simulation in engineering simulators. Today we can draw a quite different picture, Fig. 3. The boundaries have almost vanished. Current computers allow the use of highly sophisticated models even in real-time applications and CRT based control rooms are eliminating the major difference between training simulators and plant analyzers. The major difference between a training simulator and a plant analyzer can be that the models of the analyzer must be very flexible and easy to modify. However, if training does not take all the capacity of a modern simulator, it can very well serve as a plant analyzer too. We have a lot of experience in this respect: VTT’s researchers have spent many nights and weekends at the Loviisa training simulator. A strong current trend is the need for more efficient use of plant analyzers: users are no more supposed to be experts in computer and simulation technology, they should be allowed to concentrate on understanding the process being studied.

The first CSNI specialist meeting on simulators and analyzers in summer 1992 attracted 86 participants from 12 countries. In this second meeting, we have 92 participants from 17 countries. This certainly indicates a real need to gather around this theme. In addition to the large number of countries present, these meetings have another distinct, very valuable character: the participants come from virtually all types of organisations involved in the nuclear power business. From the NEA viewpoint, we bring together experts working in several principal working groups of the CSNI. I wish to thank the programme committee for putting together a strong and comprehensive programme.

I wish you a lot of exciting interactions during both the sessions and the more informal events during this week. I would not be surprised if after the panel session on Thursday afternoon we are ready to recommend a third meeting - which would mean that we have a tradition and, also, it would be time to invent a clever acronym for the meeting!
NUMERICAL SIMULATION OF DYNAMIC INDUSTRIAL PROCESSES
DEVELOPMENT OF APPLICATIONS AT VTT

1970's
- Safety analysis
- Analog & hybrid machines

1980's
- Training simulators
- Combustion power plants

1990's
- Paper mills
- Design of advanced plants
- Improvements in existing plants (cutting emissions, increasing efficiency)
- Training simulators
- CFD
- VR
COMPARISON OF DIFFERENT POWER PLANT SIMULATOR APPROACHES

Accuracy of physical description

Comprehensiveness of plant simulation
POWERS PLANT SIMULATOR APPROACHES

Accuracy of physical description

User convenience

Process analysis codes

Plant analyzers

Training simulators

Comprehensiveness of plant simulation
SUMMARY OF THE 1ST SPECIALIST MEETING ON SIMULATORS
AND PLANT ANALYZERS HELD IN LAPPEENRANTA

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ABSTRACT

The first CSNI specialist meeting on simulators and plant analyzers was an attempt to find a new perspective to simulation of the nuclear power plant processes at a time when analytical calculation and operator training were quite much diversified. The speed of computers was limited for real time physical simulation but quite impressive computer graphics and simulator development tools had been developed. Systematic methodologies for simulator validation were under development. The human reliability was under study. This paper first describes the content and conclusions of the meeting. After that the visions presented in the meeting are compared to the development during past years.

1. Introduction

The first CSNI Specialist Meeting on Simulators and Plant Analyzers was held June 9 - 12, 1992, in Lappeenranta, Finland was sponsored by the Committee on the Safety of Nuclear Installations (CSNI) of the OECD Nuclear Energy Agency (NEA). The meeting was organised in collaboration with the Technical Research Centre of Finland (VTT) and Lappeenranta University of Technology (LTKK).

The general objective of the Specialist Meeting was to exchange experiences, views and information among the participants, and to discuss the status and the future of simulators and plant analyzers. The table 1 summarised general aspects around the meeting. It is an activity report for OECD countries and companies in participating the meeting (Prt = number of participants), taking part in the meeting preparations (PC = participation in the Program Committee), giving a presentation in a session (1A, 1B, II, III, IV = session) and taking part as a panelist (Pn). The content of the meeting was defined by the Program Committee and all papers were invited.

The Specialist Meeting was structured into five sessions and a panel discussion. In the first sessions the 1A: Objectives, requirements and 1B: Concepts of simulators were discussed against present standards and guidelines. The session II: Analytical models focused on the capabilities of current analytical models, and the session III: experiences gained so far from the applications. The last session IV: Future concentrated on simulators currently under development and future plans. A panel discussions was arranged on the last day. The table 1 summarised the activities form different countries. Altogether 40 papers were included in the meeting. The list of papers is included in the Appendix.

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<td></td>
<td>INPO</td>
<td>1</td>
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<td></td>
</tr>
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<td></td>
<td>NUMARC</td>
<td></td>
<td></td>
<td>1</td>
<td></td>
<td>1</td>
<td>1</td>
<td></td>
<td></td>
</tr>
<tr>
<td></td>
<td>SCS</td>
<td>1</td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td>1</td>
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<tr>
<td></td>
<td>USNRC</td>
<td></td>
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<td>1</td>
<td></td>
<td></td>
<td></td>
<td>1</td>
</tr>
<tr>
<td></td>
<td>S3-Technology</td>
<td></td>
<td></td>
<td></td>
<td>1</td>
<td></td>
<td></td>
<td></td>
<td>1</td>
</tr>
<tr>
<td>OECD</td>
<td>NEA</td>
<td></td>
<td>1</td>
<td>1</td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>Total</td>
<td></td>
<td>86</td>
<td>10</td>
<td>4</td>
<td>6</td>
<td>12</td>
<td>9</td>
<td>9</td>
<td>6</td>
</tr>
</tbody>
</table>

Table 1. Contribution from different countries and companies in the first simulator meeting
2. Simulator concepts

For different types of training tasks it is essential, what kind of simulators serve best the various phases of the training cycle. A clear classification (1) was proposed for full scope / partial scope and basic principle simulators. A subdivision categorises the simulators into plant specific and generic simulators. The full scope simulator is a operator training tool, basic principle simulator includes physical model without real operator interface and partial scope simulator describes only a limited part of the process.

The presentations related to different type of simulator concepts are summarised in table 2. The “Nr” column is referred to the paper index in the Appendix.

The experiences on the THANGE Multi-Function Simulator indicated that the size of current standard CRT screens and the delays in swapping images cause serious limitations. The modern graphical workstations with large screens were seen acceptable as softpanel CRT’s.

The Sizewell B full scope simulator was an excellent example on an attractive, personal solution for the hardware configuration. The system uses parallel processing in 43 microprocessors, (own processor e.g. for core, primary system and steam generator). The advanced features in the application are the flexibility for the further development and possible extensions.

The SIPA simulator pioneered in implementation of advanced simulation models, CATHARE-SIMU into a simulator, which also included an advanced user interface for teaching the two-phase details. The plant modelling Man Machine Interface System is constituted by the SWORD tool.

The full scope simulator of the IKATA 3 N.P.P. was a full replica includes possibility for steady state, normal operation and Design Basis Accidents (CANAC-code). Detailed modelling is regarded for the Control Rod Driving System and 3D neutron dynamics. 3D neutron dynamics is displayed.

The Monju simulator included a full replica control room for the FBR type natrium cooled reactor.

RELAP5-based NPA included tools for calculation visualisation. Because the RELAP5-code has pioneered in the thermohydraulic simulation, also the NPA interface could be considered as such. The software includes tools for creating online graphics on the nodalization level. Also to day the system is not very user friendly and special affection to the RELAP5 world is needed that the is used extensively.

ATLAS plant analyzer using ATHLET code served as a multi-purpose engineering analysis tool for transient and accident analysis, testing of operational and emergency procedures, and evaluating of computerised control room supporting tools.

An interactive simulator-based education system was used for studying BWR emergency procedure guidelines (EPG), EPG-ICAI (Intelligent Computer Aided Instruction). This was an example on the computerised education systems.
<table>
<thead>
<tr>
<th>Nr</th>
<th>Simulator</th>
<th>Type</th>
<th>Comment</th>
</tr>
</thead>
<tbody>
<tr>
<td>4</td>
<td>TINHANGE</td>
<td>Softpanels in FSS</td>
<td>Practical for multifunction simulator, plant conformity recommended, response speed essential</td>
</tr>
<tr>
<td>5</td>
<td>SIZEWELL</td>
<td>PWR FSS</td>
<td>Marconi concept with 43 parallel computing microprocessors</td>
</tr>
<tr>
<td>6</td>
<td>SIPA</td>
<td>Advanced, CATHARE based</td>
<td>Used for training, studies and analyses. Thermohydraulics visualisation, e.g. bubble motion visualisation</td>
</tr>
<tr>
<td>7</td>
<td>IKATA 3</td>
<td>PWR FSS</td>
<td>PWR with advanced control board</td>
</tr>
<tr>
<td>9</td>
<td>MONJU</td>
<td>FBR FSS</td>
<td>Simulator standard ANSI-3.5 criteria applied for FBR, 282 malfunctions validated with reference code, modular component based modelling</td>
</tr>
<tr>
<td>10</td>
<td>RELAP-NPA</td>
<td>Engineering simulator</td>
<td>RELAP5 analysis code equipped with graphical interface</td>
</tr>
<tr>
<td>32</td>
<td>ATLAS</td>
<td>Nuclear plant analyzer</td>
<td>ATHLET based NPA with graphical visualisation and control facilities. An analysis decision support (ADS) for procedure presentation and analysis, to get knowledge on PSA, Diagnostic and explanation, Background information.</td>
</tr>
<tr>
<td>35</td>
<td>EPG-ICA1</td>
<td>Interactive education system</td>
<td>Intelligent Computer Assisted Instruction system for creating BWR plant operators Emergency Procedure Guidelines</td>
</tr>
<tr>
<td>37</td>
<td>IAE-project</td>
<td>Super-simulator</td>
<td>A challenging plan to create a supersimulator in 15 years with 3D neutronics, advanced thermohydraulics and turbulence.</td>
</tr>
</tbody>
</table>

Table 2. Simulator concepts presented (FSS = full scope simulator)

The plans for the IAE super-simulator were most challenging including severe accidents, 3D multimedia interface, exceeding real time simulation and innovative algorithm for various simulation models.

<table>
<thead>
<tr>
<th>Nr</th>
<th>Simulator</th>
<th>Hardware realisation</th>
</tr>
</thead>
<tbody>
<tr>
<td>17</td>
<td>Technatom PWR / BWR</td>
<td>TRACG-based simulator on CRAY X-MP 14 vectorial computer</td>
</tr>
<tr>
<td>25</td>
<td>ATLAS</td>
<td>AMDAHL 5870</td>
</tr>
</tbody>
</table>

Table 3. Simulator hardware realisation, applications using mainframe computers

The computer development has been an essential conditions for the simulator model development. But in this development five years is too long and the computer concepts presented five years ago are not interesting any more. In the table 3 only two hardware concepts are presented based on the
mainframe computers. At present the supercomputers have replaced mainframe computers. Because the need for the thermohydraulic and neutronics calculation needs have not increased to the level of supercomputers, the workstation based systems may gradually satisfy the computation need.

The user interface is one part in creation of the modern simulation surroundings. In table 4 the concepts for realisation the user interface in different type of simulators are presented.

The IKATA 3 represented the most modern full scope simulator concept at 1992.

The PLEVIS-concept offered a compact response to training of operators and engineers needs, not requiring control room replica or extensive and detailed plant modelling.

RELAP5 interface represents thinking, that the application are living all the time. The standardised displays may not be satisfactory.

The interactive graphic simulator SGI was an example on the computer system as an educating tool.

<table>
<thead>
<tr>
<th>Nr</th>
<th>Simulator</th>
<th>Topics</th>
<th>Comment</th>
</tr>
</thead>
<tbody>
<tr>
<td>7</td>
<td>IKATA 3</td>
<td>FSS interface</td>
<td>Control board system with 11 CRT, instructor console with 2 CRT, local operation console with 1 CRT touch panel + 4 CRT, 20 educational screens</td>
</tr>
<tr>
<td>8</td>
<td>PLEVIS</td>
<td>Graphical ABWR simulator</td>
<td>Real time plant engineering simulator, process diagrams, instructor systems, logic display, parameter graph</td>
</tr>
<tr>
<td>10</td>
<td>RELAP5-NPA</td>
<td>Analysis visualisation</td>
<td>Possibility to build an educative physical visualisation based on nodalization</td>
</tr>
<tr>
<td>35</td>
<td>SGI</td>
<td>Interactive education system</td>
<td>Six 19” colour monitors for instructor systems, panel simulation, SPDS, plant computer displays, system visualisation, variable representations. The training system may be used for the whole personnel.</td>
</tr>
</tbody>
</table>

Table 4. User interface realisation

3. Simulator models

The real plant systems especially in auxiliary processes and control systems includes so many components and similar systems, that the software generation tools are essential in speeding the work. In table 5 the software generation tools are summarised.

The SWORD was an efficient tool for creating process and control networks into the SIPA simulator.

The CETRAN models included in addition to the process network and electrical system the modelling of the main loop thermohydraulics as well.
The ROSE provided the software for the generation, integration and validation the models for process networks, electrical distribution and control systems control systems. The graphics in the ROSA system could be considered even an artistic product. The tools for modelling the main thermohydraulics and reactor core were not yet included.

<table>
<thead>
<tr>
<th>Nr</th>
<th>Simulator</th>
<th>Hardware realisation</th>
</tr>
</thead>
<tbody>
<tr>
<td>20</td>
<td>SWORD</td>
<td>Simulator model generator working in six steps: model generation, model assembly generation, external environment generation, simulation application generation, simulation configuration generation, run the simulation configuration. Integates HYTHERNET for hydraulic, thermal and chemistry and CONTRONET for control systems</td>
</tr>
<tr>
<td>21</td>
<td>CETRAN</td>
<td>Simulator technologies in a common simulator platforms, graphical instructors station, data base program, THLF for process networks, ELNET for electrical networks, NSSS for thermohydraulics</td>
</tr>
<tr>
<td>31</td>
<td>ROSE</td>
<td>Graphical system for process networks, electrical distributions and control systems. Man-machine interface includes an object editor, a model editor and a runtime environment. Impressive, artistic displays</td>
</tr>
<tr>
<td>36</td>
<td>APROS</td>
<td>Software system for creating simulation models for thermohydraulics in main and process systems, neutronics, automation and electrical models for nuclear and conventional processes</td>
</tr>
</tbody>
</table>

Table 5. Simulator system configuration tools presented

APROS system provides tools, solutions, algorithms and generic components for use in different simulation systems for design, analysis and training purposes. The applications included both fossil and nuclear power plants with automation and electrical systems.

Because the meeting was arranged by the thermal hydraulic task group of CSNI, the thermohydraulic models are essentially interested. The main features of thermohydraulic models are presented in table 6.

The SIPA simulator was the pioneer by applying analysis model (CATHARE-SIMU) for thermohydraulics. The original CATHARE has been speeded up for the real time simulation, and five years ago real time computation by 0.5 s timesteps was possible. This is presumably too long for rapid transients. Because the physical models can be conserved from the original code, no independent validation is needed for the simulator code.
<table>
<thead>
<tr>
<th>Nr</th>
<th>Simulator</th>
<th>Code</th>
<th>Comment</th>
</tr>
</thead>
<tbody>
<tr>
<td>6</td>
<td>SIPA</td>
<td>CATHARE</td>
<td>Simulator code based on analysis code CATHARE has been speeded-up. No</td>
</tr>
<tr>
<td></td>
<td></td>
<td>-SIMU</td>
<td>independent validation needed.</td>
</tr>
<tr>
<td>7</td>
<td>IKATA 3</td>
<td>CANAC II</td>
<td>4-equation TH-model, drift flux</td>
</tr>
<tr>
<td>11</td>
<td>SAF</td>
<td>TRACAS</td>
<td>Fast running, 5-equation drift flux model, Jacobian matrix for time</td>
</tr>
<tr>
<td></td>
<td></td>
<td></td>
<td>integration, no independent validation</td>
</tr>
<tr>
<td>12</td>
<td>PWR and BWR</td>
<td>SMABRE</td>
<td>5-equation drift flux mode, used for analysis and thermohydraulics in</td>
</tr>
<tr>
<td></td>
<td>simulators</td>
<td></td>
<td>several simulators, independent validation</td>
</tr>
<tr>
<td>13</td>
<td>PWR and PIUS</td>
<td>TRIP</td>
<td>3-equation homogeneous model for macro components, lumped parameters, no</td>
</tr>
<tr>
<td></td>
<td></td>
<td></td>
<td>independent validation, transient code</td>
</tr>
<tr>
<td>14</td>
<td>West-h. PWR</td>
<td>ACSL</td>
<td>Simulation language describing thermohydraulics, neutronics and control</td>
</tr>
<tr>
<td></td>
<td></td>
<td></td>
<td>systems, no independent validation, loops lumped together</td>
</tr>
<tr>
<td>17</td>
<td>TECHNATOM</td>
<td>TRACS</td>
<td>Simulation code developed from two-fluid model TRAC. For validation plant</td>
</tr>
<tr>
<td></td>
<td>PWR, BWR</td>
<td></td>
<td>data or results with analysis codes TRAC-PF1 (PWR) or TRACG (BWR) are</td>
</tr>
<tr>
<td></td>
<td></td>
<td></td>
<td>used.</td>
</tr>
<tr>
<td>18</td>
<td>SIPA, PWR</td>
<td>CATHARE</td>
<td>Speed-up of CATHARE-based simulator model has made real time simulation</td>
</tr>
<tr>
<td></td>
<td></td>
<td>-SIMU</td>
<td>possible, at least by 0.5 s timestep. 33 plant specific acceptance tests</td>
</tr>
<tr>
<td></td>
<td></td>
<td></td>
<td>are defined.</td>
</tr>
<tr>
<td>21</td>
<td></td>
<td>NSSS</td>
<td>5-equation drift flux model, no independent validation</td>
</tr>
<tr>
<td>22</td>
<td>PWR and BWR</td>
<td>ANTHEM</td>
<td>5-equation drift flux model, validation against SONG and FIST tests,</td>
</tr>
<tr>
<td></td>
<td>simulators</td>
<td></td>
<td>elegant visualisation displays</td>
</tr>
<tr>
<td>25</td>
<td>ATLAS</td>
<td>ATHLET</td>
<td>5-equation drift flux mode, initially developed for analysis application,</td>
</tr>
<tr>
<td></td>
<td></td>
<td></td>
<td>independent validation program. In the application used for BWR plant</td>
</tr>
</tbody>
</table>

Table 6. Thermohydraulic models for main systems

The use of TRACS was another example on the use of an advanced model in the plant specific simulator. The code was a speeded version of the analysis code TRAC (TRAC-PF1 for PWR and TRACG for BWR).

The ATLAS nuclear plant analyser was the third example on the simulator using advanced code, here ATHLET.

But in general most of the simulator codes were developed only for the simulator application and tested only against plant transient were most typical five years ago. With these codes the assessment
against separate effects tests could not be performed. The codes TRIP, CANAC II, ACSL and NSSS belong into this category.

But there is also a third category of the codes, which are not internationally well known, which are developed for the real time computation, but which include quite much validation efforts as well. The codes ANTHEM and SMABRE were based on the 5-equation thermohydraulics and the phase separation was solved by the drift flux mode. The validation of codes included separate effects and integral tests. In simulator application the codes was validated against real plant data.

The core neutronics models were not extensively presented in the meeting. Two models presented are summarised in table 7. One reason may be that the Program Committee was thermohydraulics oriented. Another reason may be that at the time of the meeting the real time simulation was possible only by simplified models.

<table>
<thead>
<tr>
<th>Nr</th>
<th>Simulator</th>
<th>Code</th>
<th>Comment</th>
</tr>
</thead>
<tbody>
<tr>
<td>7</td>
<td>IKATA 3</td>
<td>-</td>
<td>2-group diffusion model, 30 axial mesh points, 3D neutron dynamics</td>
</tr>
<tr>
<td>19</td>
<td>PWR, PHWR (CANDU)</td>
<td>COMET</td>
<td>3D, multinodal core model, based on equivalent theory, combines fine mesh for accurate analysis and coarse mesh for real time analysis, e.g. for PWR 38720 fine mesh points and 108 coarse mesh points. Validation: LMW benchmark. Application: load follow, rod drop, rod eject</td>
</tr>
</tbody>
</table>

Table 7. Reactor core neutronics models

The core model IKATA 3 model included quite large nodalization and the model is described as a dynamic 3D model. The solution method was not described in detail.

The a fully dynamic nuclear reactor kinetics model (COMET) was an example on effort done on core physics to fulfill simulators real time computation requirement but at the same time be able to 3D analyses. The model was validated by a series of benchmarks, and was applied successfully to existing (PWR and PHWR) configurations.

The neutron physics in many simulators was more developed in simulators and plant analyzers than in conventional T/H codes, because 1D axial, dynamic solution and some level 3-dimensionality were a minimum requirement in many simulators.

There was also independent, transportable models also for other sections of the system. The development for severe accident models had started as well. These models are described in table 8.

The description of fuel rod analysis code (TRANSURANUS) offered one possibility for the fuel rod modelling. This code is a quasi 2-D code and is ready to be implemented into simulators.

The Eurosim compact simulator used the SMABRE code for thermohydraulics of the reactor system and SCOVE for the containment. The JAPC simulator was upgraded by new feature
including among others the treatment of noncondensable gases, hydrogen and oxygen generation (radiolysis, metal, water, reactions).

<table>
<thead>
<tr>
<th>Nr</th>
<th>Simulator</th>
<th>Code</th>
<th>Comment</th>
</tr>
</thead>
<tbody>
<tr>
<td>6</td>
<td>SIPA</td>
<td>PAREO 9</td>
<td>Containment models, with hydrogen planned to implement</td>
</tr>
<tr>
<td></td>
<td></td>
<td>JERICHO</td>
<td></td>
</tr>
<tr>
<td>6</td>
<td>SIPA</td>
<td>ESCADRE</td>
<td>Severe accident model planned</td>
</tr>
<tr>
<td>7</td>
<td>IKATA 3</td>
<td>POL</td>
<td>Control system model (digital and analogue elements)</td>
</tr>
<tr>
<td>7</td>
<td>IKATA 3</td>
<td>NODE/LIN</td>
<td>Process network model (tank, heater, condenser, valve, pump, turbine ...)</td>
</tr>
<tr>
<td>15</td>
<td></td>
<td>Transuranus</td>
<td>Fuel rod thermal and mechanical analysis code, transportable, validated against experimental data</td>
</tr>
<tr>
<td>28</td>
<td>JAPC</td>
<td>SCOVE</td>
<td>New items for noncondensibles, hydrogen and oxygen by radiolysis, metal water reaction, containment flammability.</td>
</tr>
<tr>
<td></td>
<td></td>
<td>SMABRE</td>
<td></td>
</tr>
<tr>
<td>29</td>
<td>Halden</td>
<td>DISKET</td>
<td>Halden project is a pioneer in developing new computerised systems. DISKET = rule based system for diagnostics of malfunctions, SCORPIO = core surv. system, CYGNUS = core simulator</td>
</tr>
<tr>
<td></td>
<td></td>
<td>SCORPIO</td>
<td></td>
</tr>
<tr>
<td></td>
<td></td>
<td>CYGNUS</td>
<td></td>
</tr>
<tr>
<td>40</td>
<td>ARTSAS</td>
<td>Severe</td>
<td>A challenging plan for a severe accident code. It aimed</td>
</tr>
<tr>
<td>(S3)</td>
<td></td>
<td>accident</td>
<td>for accident management, individual plant examination, train in severe accident mitigation. It simulates radioactivity, noncondensable gas, clad ballooning, flow blockage, core melt, core relocation, corium progress, core refill, hydrogen generation.</td>
</tr>
<tr>
<td></td>
<td></td>
<td>model</td>
<td></td>
</tr>
</tbody>
</table>

Table 8. Models for other systems than thermohydraulics and neutronics

One trend was promises on future development for severe accident software. Application of ESCADRE and ARTSAS in simulators were planned.

The Halden experiences emphasised the importance of pre-diagnose and knowledge basis tools as support to operators work, feedback coupling to measurements and calculations (application of 3-D core) coupled with a good measurement system.

4. Simulator experiences

One impression was that the simulation models were grown and developed during application, partly because of weak points are detected during application, partly because requirements with regard to extent and fidelity of simulation have increased with application. This kind of process may be considered as a natural development.

But of course the basis work is needed for testing the simulators. For simulator acceptance tests there were in some companies quite extensive procedures. The simulator testing procedures were discussed in some presentations and the topics discussed are described in table 9.
<table>
<thead>
<tr>
<th>Nr</th>
<th>Simulator</th>
<th>Comment</th>
</tr>
</thead>
<tbody>
<tr>
<td>16</td>
<td>DOEL, TIHANGE</td>
<td>Well defined simulator development steps and systematic data collection steps defined: 1. Technical specification, 2. Plant data collection, 3. Software development, 4. Separate and integral software testing, 5. Acceptance and qualification, ANSI-3.5 followed, comparison with RELAP5-results</td>
</tr>
<tr>
<td>17</td>
<td>TECHNATOM TRACG</td>
<td>TRAC-based simulator code TRACG tested against results calculated with TRAC-PF1/MOD1. For transients validation against plant data</td>
</tr>
<tr>
<td>30</td>
<td>Fermi</td>
<td>BWR-6 simulator upgraded successfully with the CETRAN-based platform</td>
</tr>
</tbody>
</table>

Table 9. Simulator testing procedures discussed

The testing procedure for DOEL and TIHANGE simulators could be considered as one of most systematically defined procedure. It was like an application guideline to the ANSI-3.5 standard.

The validation of the TRACG simulator against TRAC-analytical model and plant transient could be considered as an example on careful simulator validation.

The Ferri upgrade is an example on implementing new process network, controller and thermohydraulic software into existing simulator. The advances in using the advanced software platform for the purpose were clearly seen.

The experiences in simulator application were very useful guidelines for the future simulator projects. The simulator experiences were discussed distinguished in many presentations and the table 10 summarised the experience profile presented in the meeting.

The multipurpose multiprocessor analyzer SAF was used for basic training of engineers, system design and optimisation, special safety studies such as long term phase of a small LOCA, development and validation of operating procedures and the development of emergency scenarios.

With ATLAS numerous sensitivity studies were carried out. With a course nodalization model much faster than real time.

In presentation the INPO experiences it was emphasised that he integration of plant and industry experience in the training program is extremely valuable. Self critique is an effective means of improving the performance of an operating team. In real life the plant operators experience very infrequently even most common operational transients, including shutdowns and start-ups. In future plant designs vendors are proposing the use of modern supervision and control technologies that automate start-up, shutdown and the control of many operational transients.
<table>
<thead>
<tr>
<th>Nr</th>
<th>Simulator</th>
<th>Topics</th>
<th>Comment</th>
</tr>
</thead>
<tbody>
<tr>
<td>2</td>
<td>KWU, FSS</td>
<td>Training content</td>
<td>Theoretical, practical and simulator training</td>
</tr>
<tr>
<td>3</td>
<td>NUMARC</td>
<td>Training trends</td>
<td>New focus, Human factors research, future trends</td>
</tr>
<tr>
<td>23</td>
<td>EDF PWR</td>
<td>Facilities, programs</td>
<td>Training combines 1. Basic principle and part task simulators, full scope simulators, multifunction simulator and SIPA simulator.</td>
</tr>
<tr>
<td>24</td>
<td>SAF</td>
<td>Plant analyzer</td>
<td>Several tasks for the multipurpose simulator: basic training, validation accident procedures, emergency drills</td>
</tr>
<tr>
<td>26</td>
<td></td>
<td>INPO experiences</td>
<td>1. Plant and industry operating experiences important, 2. Self critics by crew members needed, 3. Site specific simulators effective</td>
</tr>
<tr>
<td>27</td>
<td>Essen, PWR III</td>
<td>Accident management</td>
<td>After developing emergency procedures for German PWR plants, the personnel was oriented into the procedure by using the simulator. Training was a success, why not.</td>
</tr>
<tr>
<td>33</td>
<td>TEPCO</td>
<td>BWR and ABWR simulators</td>
<td>Current training combines 1. Full scope simulator, 2. Compact simulator. Problems: training frequency, little troubles in plants, more complex procedures with advanced automation</td>
</tr>
<tr>
<td></td>
<td>SA simulation</td>
<td>management</td>
<td></td>
</tr>
<tr>
<td>39</td>
<td>SCS opinion</td>
<td>Severe accident</td>
<td>Development has started for implementing severe accident modules into simulators. But real plant measurements can give only limited information on existing state.</td>
</tr>
<tr>
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Table 10. Training experiences discussed

In Germany (Essen) accident management measures (primary and secondary feed and bleed) were implemented in existing PWRs. Consequently training of these cases in a full scope simulators was necessary.

The general impression obtained from the presentation was that the simulator applicants are quite satisfied with the performance of their simulation tools. Is this the right impression? Hopefully not, because such a satisfaction will not create motivation for further development.

The human performance was considered very important in simulators. The human performance was not essential only by operators, but also by engineers.
Plant specific full-scope simulators were not judged to be absolutely necessary, due to the high degree of automation in German NPPs. The Part Task simulator with comprehensive modelling of all safety relevant systems available at KWU training centre augments efficiently the overall simulator training. In training theoretical knowledge and simulator courses are essential.

5. Visions - did they come through

In many presentations and in panel discussion many vision for the future development were presented. Here some opinions are discussed, especially against the development during past years. The questions discussed in the panel included:
Are simulator good enough?
Is assessment adequate?
How handling human factors?
What kind of standards are needed?
Is international assessment and standard problems needed?
Where are limits of simulations?
What is future of simulator codes with respect to advanced best estimate codes?
How operators recognise simulator limits?

It was seen that future trends include evolution of physical models and numerical algorithms for simulators and plant analyzers. Extensive validation efforts are needed to ensure the similarity with analysis tools. The physical models will also address phenomena which are important for severe accident management, both prevention and mitigation. The advanced simulators will be used e.g. for training plant personnel, for design studies involving new reactor systems, and for validation of accident management procedures. This vision has been realised. The simulator models and analysis models are approaching. It is waste of time developing separate models for simulators and plant analyses. The fast development of computers supports this development.

What is future of simulator codes with respect to advanced best estimate codes? The development has shown, that the gap is narrowing.

In a speech of training centre personnel it was indicated that the human factors are essential in the operator training. A special team is needed for studying human problems. Adding multimedia effects will increase. Multimedia will surely have continuously important role in developing training methods. In 1992 multimedia was understood as a locally implemented systems giving improved visual and sound effects. The operator should live like in the real control room. The present multimedia development gives more possibilities to this.

It was seen important to transfer the know-how and competence of the generation who built the nuclear power plants to the present generation operating the plants. Low power and midloop operations are coming in the near future. It may be asked if the old generation understands the physics better than the new generation, using advanced simulation model.

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Cooperation between OECD/CSNI and SCS was wished. It was believed that CSNI understands well safety analyses with detailed models and SCS the commercial simulation. This cooperation has not perhaps been realised. One reason may be, that the pure commercial nature from the simulation is withdrawing and speed-up of computers simply make application of analysis codes possible. Gap between simulators and plant analyzers has narrowed. In this discussion a proposal for international simulator related benchmarks was presented. Due to plant specific character of simulator it is clear, that the benchmarks are possible only for transportable codes with flexible input generation.

It was recommended that mental model of operators should be developed. The problem may be, that the creative thinking and learning process is difficult to be predicted in the mental models. Human beings are no machines.

It was seen that safety support systems are needed for supporting the operators. This is surely true, but it takes some time before the operator rely on computerised support tools.

There was an attempt to systematise the terminology for full scope, basic principle, partial scope, plant specific, generic and engineering simulators. The term are self explaining, but the present simulators are not easily categorised into these classes. Perhaps the other terms than the full scope give too simplistic imprint on the simulation.

A good co-operation between different experts, e.g. thermohydraulists, neutron physicists, human factor specialists, plant engineers and operators was recommended. This has perhaps been realised.

Trend towards multifunction simulators and NPA’s was send for training operators, technical staff, design and safety analyses. Soft panels lower cost alternative for multifunction simulator. These trends have sure been realised. Also control rooms may change completely.

Operators may rely too much on the system? This question has to be discussed continuously.

Speeding of codes for NPA needed. Problems in combining TH and neutronics models was realised. This studies have been started inside CSNI as a separate task.

The task for super-simulator development was challenging. It may be that the realisation has been slower than expected.

Simulation to be extended to other transients scenarios, in the case of instability incidents, an extension of models scope is necessary. The support in this area from CSNI side could be needed.
Appendix: Papers presented in the First Simulator Meeting, 9 - 12 June, 1992

IA: Objectives, requirements
1. Classification and Optimization of Training Tools for NPP Simulator, G. VAN BILLOEN, Tractabel, Belgium
2. Role of Simulators in Licensing and Operator Qualification in Germany, D. GRONAU, A. KUHN, Siemens, Germany
3. Role of Simulators in Licensing and Qualifications of Operators R. C. EVANS, Numarc, USA
4. A Utility Perspective on the Need and the Possibility of Diversifying the Use of the TIHANGE 2 Full Scope Simulator G. CLERMONT, Tractabel, Belgium

II: Concepts
5. The United Kingdoms First PWR Full Scope Training Simulator, J. A. RATCLIFF, Nuclear Electric, UK
6. SIPA Simulator: Objectives of EDF and IPSN - Further Developments, J. PELTIER, F. GRIMALDI, CEA, France
7. The Most Up-to-date Simulator for the P.W.R. Advanced Control Board, Y. NISHIMURA, T. TSUKAMOTO, Mitsubishi, Japan
8. EWS Based Visual and Interactive Simulator for Plant Engineering, S. OHTSUKA, K. TANAKA, E. YOSHIKAWA, Toshiba, Japan

II. Analytical models
11. Recent Analytical Model Developments for the FRAMATOME SAF Plant Analyser, J-P. BALEON, L. DZIALOSZYNKI, J-L. GANDRILLE, Framatome, France
12. Modelling of Thermalhydraulics and Reactor Physics in Simulators, J. MIETTINEN, VTT, Finland
13. A Simplified Model for the Simulation of Operational and Accident Transients of PWRs and PUS Reactor, E. BREGA, C. LOMBARDAI, M. RICOTTO, A. RILLI, ENEA, CeSNEF, Italy
14. A Simulation Program for Westinghouse PWR's, G. MARELLA, R. CALVO, ENEA, Italy
15. Westinghouse, USA
16. TRANSURANUS: A Fuel Rod Analysis Code Ready for Use in Simulators and Plant Analyzers, K. LASSMAN, ITE, Germany
18. Assessment of Training Simulators with Advanced Models, R.M. FANEGAS, J. J. MUNOZ, J. A. RUIZ, A. TANARRO, Technatom, Spain
19. Methods Employed to Speed up CATHARE for Simulation Uses, J.-M. AGATOR, IPSN, France
20. Validation of COMET - reactor Core Model based on Equivalence Theory, C. PAUQUETTE, H. TRANDUC, F. FRIEDMAN, CAE, Canada
21. Automatic Creation of Simulation Configuration the SIPA Workshop: SWORD, G. OUDOT, A. VALEMBOIS, Thomson, France
22. Implementation of Thermal Hydraulic Network Simulation Programs in CETRAN Environment, D. TRUMPLER, I. ALATALO, ABB, USA, ABB, Finland
23. ANTHEM: Advanced Thermal-Hydraulic Model for Power Plant Simulators, R. BOIRE, J. SALIM, CAE, Canada

III. Experience
23. Experience Feedback on Operator Training on Simulations at EDF, P. BILLARD, EDF, France
24. FRAMATOME Experience in the SAF Plant Analyser Operation, B. BALLOT, Framatome, France
25. BWR Transient Simulation within the German Nuclear Plant Analyzer (ATLAS), W. POINTNER, H. AUSTREGEISL, GRS, Germany
26. Experience in the Use of Simulators, M. B. LEVITAN, INPO, USA
27. Simulator Training in Accident Management - First Experiences in Germany, R. ISENBÜGEL, GSF, Germany
30. Full Scope Upgrade Project for the Fermi 2 Simulator, D. BOLLACASA, J. B. GONSALVES, P. C. NEWCOMB, ABB, USA
31. ROSA: A Realtime Object Oriented Software Environment for High Fidelity Replica Simulation, A. ABRAMOVITCH, L. WHITE, CAE, Canada

IV. Future
32. Open Design of ATLAS - Architecture and Future Extensions, D. BERAHA, T. VOGGENBERGER, GRS, Germany
34. The Interactive Graphic Simulator (SGI): A Flexible Tool for a Customized Training and Other Purpose, F. ORTEGA, A. CARDOSO, Technatom, Spain
35. An Interactive Simulator-Based Education System for BWR Emergency Procedure Guidelines, N. TANIKAWA, T. SHIDA, Hitachi, Japan
36. APROS-Based Lovisa Nuclear Power Plant Analyzer, O. TIHONEN, M. HÄNNINEN, E. K. PUSKA, VTT, Finland
37. The Innovative Simulator for Nuclear Power Plants, A. KUROSAWA, H. OHASHI, M. AKIYAMA, IAE, Japan, Tokyo University, Japan
38. Use of Simulators in (Severe) Accident Management, R. C. EVANS, Numarc, USA
39. On the Use of Simulation Tools for Beyond Design Base Accident Management and training, A. SHARON, W. A. THOMAS, SCS, USA
40. Simulation of Severe Accident in Reactor Core for Training and Accident Management (paper not presented), M.R. Fakory, A. Onyemachi, S3, USA
Human Machine Interaction
Research Experience and Perspectives
as seen from the OECD Halden Reactor Project

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Abstract

In this paper a short review is given on important safety issues in the field of human
machine interaction as expressed by important nuclear organisations such as USNRC,
IAEA and the OECD NEA. Further on, a presentation is offered of research activities at
the OECD Halden Reactor Project in the field of human machine interaction aiming to
clarify some of the issues outlined by the above mentioned organisations.

The OECD Halden Reactor Project is a joint undertaking of national nuclear
organisations in 19 countries sponsoring a jointly financed research programme under the
auspices of the OECD - Nuclear Energy Agency. One of the research areas is the man-
machine systems research addressing the operator tasks in a control room environment.
The overall objective is to provide a basis for improving today's control rooms through
introduction of computer-based solutions for effective and safe execution of surveillance
and control functions in normal as well as off-normal plant situations.

1. Introduction

There is, in particular, three nuclear organisations that have a strong and positive
international reputation of taking the problems related to human factors seriously in
terms of research programs and development of guidelines. The aim, of course, is that
nuclear control rooms are designed to good human factors engineering principles and
that operator performance and reliability are appropriately supported ensuring public
health and safety. The organisations are the United States Nuclear Regulatory
Commission (USNRC), the International Atomic Energy Agency (IAEA) and the OECD
Nuclear Energy Agency (NEA). In the following quotations from important documents
from these organisations are given to demonstrate the importance they place on the Human Machine Interaction issues.

The “Advanced Human -System Interface Design Review Guideline” authored by John O’Hara and issued by the USNRC (NUREG/CR-5908) have the following introduction to the Human-System interface problem:

“The importance of a well designed human-system interface (HSI) to reliable human performance and nuclear safety is widely acknowledged. The US National Academy of Sciences (Moray and Huey, 1988) reported that one of the first insights from studies of the Three Mile Island (TMI) accident was that errors caused by operators in the Control Room (CR) are a significant contributing factor to nuclear power plant (NPP) incidents and accidents. The errors at TMI were due to a number of factors including a poorly designed CR and inadequate provisions for monitoring the basic safety parameters of plant functioning.

The International Nuclear Safety Advisory Group (INSAG, 1988) of IAEA, in their basic safety principles, indicated that “…one of the most important lessons of abnormal events, ranging from minor incidents to serious accidents is that they have so often been the result of incorrect human action”. Further, “…continued knowledge and understanding of the status of the plant on the part of the operating staff is a vital component of defences in depth.” This conclusion led to the IAEA safety principle that: Parameters to be monitored in the CR are selected, and their displays are arranged to ensure that operators have clear and unambiguous indications of the status of the plant conditions important to safety, especially for the purpose of identifying and diagnosing the automatic actuation and operation of a safety system or the degradation of defences in depth (page 43).”

In the IAEA Safety report: “Safety Issues for Advanced Protection, Control and Human-Machine Interface Systems in Operating Nuclear Power Plants” (IAEA, 1997) one chapter deals with safety approaches for human-machine interfaces. In the introduction it is stated: “…The human-machine interaction problems are complex. In many applications, the role of the human operators is often neglected in design and the human functions are defined by default, governed by the limitations and gaps of hardware and software. It is questioned whether the role defined by implication for the human operator can be effectively and reliably performed. For example:

• Is information presented at a sufficiently high level that it supports human decision making?
• Does information integration cause additional cognitive burdens?
• Are displays easily readable and understandable?
• Is information readily accessible?
Operators are often conservative and reluctant to accept technology changes. To avoid problems with user acceptance, verification and validation program has to be prepared to ensure adequate testing of the new human-machine interface. Dynamic testing and validation utilising training simulators may reveal possible problems long before plant installation. The installation of new systems and ways to work has influence on the whole organisation and should be carefully evaluated."

OECD NEA has established a Senior Group of Experts on Safety Research (SESAR) with members having wide responsibilities and experience in OECD Member countries’ nuclear power programmes. The task of the SESAR group is to review current situation in Member countries with regards to safety research, to reflect on a rational for safety research in the years to come, to identify future needs, and to establish a priority list (NEA,1993).

In the 1993 report it is stated on human factors (page 23): .."Humans have a vital role to play in all aspects of normal operation and their performance in the role can determine the safety of the activity. Additionally the flexibility and ability of humans can contribute significantly to the successful management of accident situations which have developed. In all these aspects the human factor has become of increased importance. The topic embraces man/machine interfaces and communication in the control room and other plant areas.; the use of computers and the reliability of the associated software, the role of simulators in training and simulation exercises; efficient and effective maintenance and its quality assurance; total safety management; system effectiveness; characterising the performance of individuals and groups in modelling the total plant safety system; and the optimisation of the balance between human activity and engineered, automatic response. It is a vast and difficult area involving interdisciplinary research from sociological, physiological, and technical areas where the use of cognitive science techniques is becoming essential in addressing human performance." In the Priorities chapter (page 31) it is stated: .." It is the opinion of SESAR that that it is impossible to exaggerate the importance of establishing an improved understanding of the performance of humans in the vast range of activities that are related to reactor safety."

In the 1995 SESAR follow-up report (NEA draft, November 1995): "Nuclear Safety Research in OECD countries, Areas of Agreement, Areas for Further Action, Increasing Need for Collaboration" some of the major themes for further research in the areas of Human Factors are:

- Characterising and assessing the performance of individuals, teams and organisations

- Man-machine interfaces and communications in the control room and other plant areas
• Selection and training of staff

and some from the closely coupled area of Plant Control and Monitoring are:

• Signal validation methodologies for severe accident situations

• Development of operator support systems using advanced data processing and human-machine interfaces

• Condition monitoring methods

The former Director of the Office of Nuclear Regulatory Research at USNRC, Dr. Eric S. Beckjord wrote a 10-year vision on NRC Research in 1995. Here he underlines that “international collaboration is vital to success and safety of nuclear installations, including operation, regulation and safety research. In the case of safety research the benefits of international collaboration are cost sharing and avoiding unnecessary duplication of effort in experimental projects, bringing the minds of best qualified people together on a world scale, and rapidly dissemination of results.”

In the 1995 SEASAR report (page 21) the OECD Halden Reactor Project is referenced as a highly successful example of international collaboration in several aspects of human factors. By close co-operation with the Halden Project even small national research programmes may be viable.

2. Human-Machine Systems Research at the Halden Project

The Halden Project has since 1970, through international co-operation, successfully conducted research and development in areas related to control room systems development and human factors.

The research programme has addressed the research needs of the nuclear industry in connection with introduction of digital I & C systems in NPPs. The programme has provided information supporting design and licensing of upgraded, computer-based control room systems, and demonstrated the benefits of such systems through test and evaluation experiments in Halden’s experimental research facility, HAMMLAB and in pilot installations in NPPs.
2.1. Human Factors Research

The aim of the human factors research is to provide knowledge about the capabilities and limitations of the human operator in a control room environment. Understanding the impact of new technology on the role and performance of operating personnel is crucial in decision making concerning safety of nuclear power plants.

At the Halden Project this understanding is accomplished through a combination of activities addressing operators cognition and information processing methods in various control room work situations, function and task allocation methods and test and evaluation of advanced support systems. In addition to providing knowledge about human performance, the program also gives results on the new methods and measures for studying human performance.

The program is divided in three activities: i) operator cognition and information processing, ii) test and evaluation and iii) methodological development

2.2. Studies of Operator Cognition and Information Processing

2.2.1. Human Error

The aim of the activities on human error is to provide improved understanding of how operators diagnose disturbances and to identify potential errors or inefficiencies which may occur in the diagnostic process. The long term goals are to produce practical knowledge for system and man-machine interface design and to achieve better modelling of cognitive errors for representation in probabilistic safety analysis.

A series of pilot studies and one major experiment have been performed to find a reliable and valid methodology to investigate human error in a control room setting. The practical insights from these studies can be found in (9) and relates to i) diagnostic strategies and styles that have been observed in single operator and team based studies; ii) the qualitative aspects of the key operators support systems, namely MMI interface, alarms, training and procedures, that have affected the outcome of diagnosis; and iii) the overall success rates of diagnosis and the error types that have been observed in the various studies.

2.2.2. Human Centred Automation

Professor Sheridan has in (10) said: "It has become evident that humans, when put in the role of monitor, supervisor and automation back-up in case of failure, may not perform.
Humans become bored and unalert during the long period when the automation does work. They may loose track of what the automation is doing or what surrounding circumstances are. They may not understand what the implications of what they may have asked the automation to do and even forget which mode they have set the automation in. As the automation becomes more sophisticated the human's mental model may be insufficiently accurate to make the necessary inferences or predictions, such that anticipating what to do becomes difficult or impossible.

A typical solution is to say: *allocate functions to the human the tasks best suited to the human, allocating to the automation the tasks best suited to it.* This is easy to say, but not so easy to do.\"  

At the Project it is acknowledged that the role of the operating crew is either intentionally or unintentionally created by the system design process. However, it is very little empirical data about the effects of different trade-offs of function and task allocation on the performance of control room crews. The Project are about to study the effect of different task allocation methods in particular with respect to analysing and presenting plant automatics.

### 2.3. Test and Evaluation

The experimental evaluation of the Computerised Operator Support Systems (COSSs) and MMIs developed at the Project forms the basis for the human factors work at Halden. The setting for this experimentation is the NORS simulator situated in HAMMLAB, see below. NORS is a full-scale simulator of a PWR plant and while not being an exact replica of any one nuclear power plant, is, nevertheless, representative of a typical PWR.

Over the years a well-established infra-structure and methodology for performing evaluation experiments of new operator aids and man-machine interfaces has been developed. Both operators from the Halden Reactor and operators from the Loviisa NPP in Finland take part as test subjects in these experiments. A variety of data can be collected during the experimental sessions: video and audio recordings of the activities in the control room, logs of all interactions between the operators and the simulator (displays used, control actions performed, etc.) as well as logs of critical process parameters of relevance for judging the performance of the operating crew during a particular transient. In addition, operator interviews, questionnaires, debriefing sessions, verbal protocols, etc. are utilised to extract additional information for the later analysis of the experiments.
The experiments performed in HAMMLAB are of different types. Some studies focus on providing direct design feedback to a specific system, while other experiments aim at providing more general knowledge for use in defining technical bases for system design and evaluation.

In addition to evaluating the final systems, there is also need for evaluations at different stages of system development to provide early feedback on the quality of certain system features. The process is therefore often iterative, running through a series of experiments on a particular system, each contributing to a better design and an improved final system.

Over the years a number of COSSs have been evaluated in HAMMLAB, (11,12). Currently, the work is focused on alarm systems, staffing level and night-shift work.

2.3.1. Alarm Systems

The need to improve alarm systems beyond the single set-point, single alarm approach has led to the development of different kinds of new alarm systems for the nuclear industry. At the Project, both a computerised alarm system toolbox - COAST (13), and a special application for HAMMLAB - CASH (14) became operational during early 1996. In the fall of 1996 CASH was used as a vehicle to carry out studies of alarm system concepts. Examples of concepts studied were: the effects of different attention getting features, the relation between alarm information and the diagnostic and problem solving process and the alarm system functioning (alarm presentation methods, structuring methods, etc.).

2.3.2. Staffing Level

The "Study of Control Room Crew Staffing for Advanced Passive Reactor Plants" was the largest test and evaluation project carried out by the Project in the 1994-96 time frame. It was a bilateral contract done for USNRC, but the results are all open and published in the Halden report series (15).

Differences in the ways vendors expect the control room staff to interact with evolutionary or advanced plants may require reconsideration of the minimum shift staffing requirements set in today's federal regulations. This research project evaluated the impact of various level of staffing on team performance. The purpose was to contribute to the understanding of potential safety issues and to provide data to develop design review guidance.

The study was conducted at the Loviisa NPP and at HAMMLAB. Loviisa served as the conventional plant while HAMMLAB served as the advanced plant. Data were collected
from eight crews during a range of design basis scenarios, each crew serving in either a normal or minimum staffing configuration. Results show that crews in the conventional plant experience less workload than crews in the advanced plant, and that minimum sized crew experience more workload than normal crews. The increased workload did not exceed the threshold beyond which performance degradation occurred as situation awareness, teamwork, and task performance demonstrated in the advanced plant were significantly better or higher for these measures than in the conventional plant.

2.3.3. Night-shift Work

Human performance is known to be greatly affected by variation in work cycles. Studies of shift-work have identified numerous incidents of peaks and troughs in human performance, in which human are at their best and worst. However, little is known how variations and work-time influences operator cognitive performance. During the next three years studies of operator performance at night will be undertaken. These studies will focus on the types of operator information processing and cognitive activities which are affected by the time of day. Experience gained from the experiments will constitute a basis for possible optimisation of shift-work and for modification of the man-machine interface, procedures or other relevant means.

2.4. Development of Technical Bases for Guideline Formulation

While guidelines for design and evaluation of control rooms using conventional man-machine interfaces are in general available today, the basic knowledge required for design and evaluation of computer-based control rooms needs to be further developed. Existing guidelines for computer-based systems mainly address the questions of how to present information (symbols, colours, font size, etc.), while little guidance is given on which information is important, and how process control should be executed in an efficient manner. Thus, existing guidelines for evaluation of computer-driven human-machine interfaces and digital based control rooms are incomplete. System and control room developers as well as organisations evaluating and licensing human-machine interfaces are therefore in need for guidelines in this field.

A major objective of the human factors research at the Halden Project is to provide experimental data which can contribute to formulation of guidelines for design and evaluation of computer-driven man-machine interfaces. A considerable part of the experiments in HAMMLAB are thus focusing on generic issues in connection with operator performance in an advanced control room environment. A key issue in this work is to find a suitable format for describing the generic results such that they easily can be applied by member organisations in formulation of their guidelines. Presently, the
Project is preparing "lessons learned" reports from the evaluation experiments performed at Halden where emphasis is placed on making the information relevant for guideline formulation available in a structured and easily accessible way.

2.5. Methodological Development

2.5.1. Situation Awareness

The situation awareness measurement technique (16) is an objective measure for studying how specific support systems assist the operator in maintaining an understanding of the status and behaviour of the nuclear plant process, especially in those case where the aim is to enhance the operator's mental representation and awareness of the process. This measure has been applied successfully in a number of studies, for example in the above mentioned staffing level project. Continued research with this measure will be used to determine whether individual crew members' situation awareness are differently developed and maintained and how shared situation awareness develops and is communicated among team members.

2.5.2. Eye-movement Tracking Measurements

Eye movement tracking measurement techniques have demonstrated the possibility of better studying operator cognitive activities and information processing. Studies in connection with the human error project have proved the usefulness in clarifying degraded verbalisation during period of greatest interest to researchers. Continued research with this technology is envisaged as part of the studies focusing on cognitive aspects of the operator.

2.5.3. Workload

There is a need to better understand the effects of workload on operator performance. Studies of workload will be carried out to collect empirical data in the control room environment, specifically the program will consider the suitability of different metrics to measure subjective workload, effects of workload transition (under- and overload) on operator performance, and the interaction with and mediation of other subjective measures of operator performance (e.g. situation awareness, verbal protocols, time and accuracy analysis).
3. Information Presentation Methods

3.1. Navigation

Use of computerised systems in the CR have resulted in increasingly complex and sophisticated workstations. Typically in such environments the control system requires operators to work with computerised systems for monitoring and control of the process.

This introduces the important requirement for operators to both orientate themselves within the workspace and be able to move around in it in order to locate information. These aspects become crucial where integration of information across different displays must take place. Whilst computerisation allows presentation of greater amounts of information in more sophisticated ways than ever before, it has also increased the potential to negatively affect operator performance. This is due to the introduction of navigation problems to the system. As upgrading of traditional and hybrid control rooms occurs issues of navigation and its impact on operator performance will be increasingly important.

Navigation itself is concerned with the determination of one's position in, and the movement through, a space. This space can be a physical, a process space (mimics), or information space (network). In computerised control rooms navigation in its simplest form is related to two crucial aspects of operator performance.

- The first is maintaining or establishing awareness of the operator's position within the network of computerised displays.

- Second is the ease and efficiency with which an operator is able to travel around within the network.

3.2. Navigation problems

A wide range of navigation issues and problems have been reported in the literature from different industries including nuclear, aviation and aerospace. For example, two important and commonly reported problems are,

1. The key-hole effect, when the work space becomes large and complex the use of VDU's allows only a partial viewing of the total process picture at any one time. This restricted key-hole has obvious and serious implications for operator performance, for example situational awareness, monitoring, problem solving.
2. The second problem is one of the operator becoming ‘lost’ or ‘disoriented’ within the information system. Here the operator loses awareness of their position within the network of displays and consequently their performance is negatively affected, i.e. failure to adopt an appropriate search strategy, increased error rate, slower search times.

Many other problems of this nature are staring to come to light as more use is made of this technology and operators find themselves having to move through increasingly sophisticated control and information spaces. At the Project a program is under way including research issues such as:

- The perceptual and cognitive aspects underpinning operators’ awareness of their orientation in, and navigation through work spaces.

- Appropriate methods for representation of work space itself in order to better support operators navigating through it.

- The development of metrics for the evaluation of navigation, both in restricted experimental settings and more sophisticated simulator environments.

- Methods for evaluation and comparison of design alternatives.

- The development of guidelines on navigation for designers of HMI systems.

3.3. Overview Displays

One of the most discussed features for evolutionary and new control rooms are so-called overview displays, one of the first examples of such a design came from Combustion Engineering through their IPSO (Integrated Process Status Overview) display. The idea is to integrate information which is normally found distributed around in the conventional control room on one permanently displayed picture where it is clearly visible to varying distances in the control room.

Project staff has designed a so-called integrated information overview display (17) where the idea is to support rapid assessment of the plant status and dynamics by a representation of the whole process. The displays is shared by the control room staff and is a strong support for co-ordination. The layout and content of the display are context dependant to match the changing operators’ needs and tasks. Centred around mimic diagrams, it combines different graphical features to support an efficient control of the
complex process. Configurable elements and an original alarm presentation support clear and rapid identification of disturbances.

The integrated information overview display in HAMMLAB 96 is today on 4 screens. 10 plant systems windows are displayed in the upper 2 screens: each window includes a chronological alarm list with room for six alarm messages, and icons at fixed locations for key alarms in the group of plant systems. Trend curves of main process parameters and availability of power are also shown in the upper screens. The lower left screen includes the reactor auxiliary & safety systems diagrams, plant protection summary and CSF summary. The lower right screen includes the water steam diagram with power balance trends.

The overview display has been implemented and integrated in HAMMLAB, it shows promising results in use and is well perceived by operators.

3.4. Alarm Displays

CASH (Computerised Alarm System for HAMMLAB) is the newly implemented alarm system for HAMMLAB. It includes several advanced alarm handling features, such as extensive alarm structuring feasibility’s and an efficient man-machine interface, aiming to reduce operator workload during plant disturbances. CASH will strongly support upcoming human factors experiments in particular the experiments on alarm systems (page 5). CASH provides two levels of presentation. At the top level, alarms are presented on the above mentioned plant-wide, high-level, system paced overview display. Here alarm information is integrated with process information rather than in a self-standing independent display.

The second level provides alarm details in detailed, totally user-controlled displays, both integrated in the NORS operating displays and in CASH selective displays. In the selective displays, which can be called up upon request from the top-field, the operator is able to look at alarms suppressed from the overview and he has access to a variety of alarm lists. By enabling a combination of the “radio-buttons” he can build the alarm list he needs for the present situation. The I/O buttons located at the bottom of the screen, enables the operator to access further information, such as trend diagrams.
4. Interactive Visual Simulation

4.1. Virtual reality

Virtual reality (VR) has been proclaimed an exciting new technology with many applications. Users of VR today reside in simulation, telerobotics, medicine, architecture, and entertainment and many other areas. The future use of VR seems limited only by the creativity of its designers.

Virtual Reality has been defined by (18) as "A computer system used to create an artificial world in which the user has the impression of being in that world and with the ability to navigate through the world and manipulate objects in the world."

Another interesting definition is: "Virtual Reality allows you to explore a computer generated world by actually being in it" (19).

4.2. Halden Virtual Reality Centre (HVRC)

HVRC is a new complementary extension to HAMMLAB. The activities in the centre are connected to on-going research at the Halden Project, but mostly to development projects for the industry such as nuclear, oil production, maritime, air-traffic control and process control. The centre expands the possibilities to develop and evaluate new systems and methods for HMI. VR is being used as a tool to explore new possibilities of process information allocation and visualisation with realistic or abstract 3D graphics in real time environments. It is used to suggest more efficient approaches to present and navigate through process displays in advanced control rooms, guide designers and inspectors in using methods for cost effective control centre design development and evaluation (Human Factors V&V); and to develop computerised training tools for outage maintenance and operational tasks.

4.3. Virtual Mock-Up's: VR as an important engineering tool in terms of improved control rooms

Across the world VR is becoming recognised as a powerful engineering tool. VR technology is today considered as an efficient solution to many industrial problems, problems that previously has been troublesome to find ways to deal with.
For example, as today's control rooms need upgrading from technological reasons there is now an opportunity to make a really good redesign by applying VR-technology. The product is a "virtual mock-up" where the end-users - the operators - , engineers and managers with assistance of human factors specialists - together can formulate the optimal solution for their new control room. The possibility to actually "walk inside" the planned control room - and consider, modify, try out, discuss, agree on the location of desks, keyboards, CRTs and control boards with their instruments - from every angle and position before any design decision has been made - is a tremendous advantage. The output from this exercise is finally a set of CAD drawings to the vendor actually building the control room.

The Halden Project is using this approach in several redesign projects for Swedish nuclear control rooms.

5. Plant Surveillance and Support systems

A common goal for both the human factors work and the systems work at the Project is to generate and test reliable and effective human-computer interfaces which can ensure operator awareness of plant situations and operating states. The work on surveillance and support systems addresses questions related to human-machine interaction in operator tasks such as fault detection, diagnosis, prognosis and procedure implementation. One emphasis is the development and tests of actual surveillance systems, another is developing man-machine interface design proposals.

The human-computer interface represents the boundary between functions allocated to operators and the software systems which are designed to support operator tasks. As such, it influences not only what information is presented to the operator, but also how it is presented, in addition to providing mechanisms for taking actions based upon decisions made about the information presented.

5.1. Core Surveillance Systems

The Core Surveillance System SCORPIO has been developed by the Halden Project to provide NPP reactor physicists and control room operators with a practical tool for improved on-line monitoring of core status and optimisation of control strategies for planned power changes. This is achieved through better surveillance of core instrumentation and application of powerful on-line core physics simulators (20).
SCORPIO operates in two modes: monitoring mode, and predictive mode. In the Monitoring Mode, the system produces a realistic estimate of the core status based on instrument readings and calculations with CYGNUS, a 3-dimensional physics model including xenon dynamics. As soon as the core state data are available, the system checks and displays the margins to operating limits.

In the Predictive Mode, the system calculates the core behaviour during a planned power transient. This is of great help for reactor operation in dynamic core state situations where xenon variations often have a complex influence in power distribution.

Thus, the operator can avoid control strategies that are unacceptable due to operational constraints, by inspecting the predicted margins to these limits for different strategies.

Usually, a predictive calculation is made by simply specifying the planned power manoeuvre through drawing the power as a function of time using the mouse device, and then starting the strategy generator, STRATOS. When the strategy generator has been run, a 3D-calculation with CYGNUS is started using the controller settings provided by STRATOS as input.

The results of the predictive calculations can be viewed during the simulator run, on a picture showing margins to operational limits. More detailed information on the forecast can be visualised on demand as trend curves or spatial core distributions in a number of pictures.

During recent years SCORPIO has been delivered to Nuclear Electric, Sizewell B NPP in UK; Duke Power, McGuire and Catawba NPPs in USA; and Vattenfall, Ringhals 2 NPP in Sweden, and during 1997 a SCORPIO version for VVER reactors will be installed at the Dukovany NPP in the Czech republic.

5.2. Accident Prevention and Management

The Project is carrying out a research programme on computerised accident management support (the CAMS-project). The aim is to establish a prototype of a system which can provide support to the control room operators and the staff in the Technical Support Centre during accident situations. The CAMS prototype utilises available simulator codes and the capabilities of computer-based tools to assist in identification of plant state, prediction of future development of the accident, and planning of accident mitigation strategies (21,22).
The first CAMS prototype consisted of a database and a knowledge base, a predictive simulator and a man-machine interface system. The system was evaluated during a national emergency drill in Sweden in May 1995 with positive results. Recently new methods and modules are added for signal validation, state identification, tracking simulation, predictive simulation, risk monitoring and the man-machine interface. This second prototype is still under development. The purpose is to demonstrate that the developed functions can efficiently work together. The further plan is to test CAMS at a power plant or a national crisis centre.

6. Halden Project HMI Activities for 1997-99

The programme for the 1997-99 period is intended to strengthen the Project’s work in the area of human factors. The work will primarily be based on a series of experiments to be carried out in HAMMLAB, for which a substantial upgrading is foreseen, thus making the facility an even stronger nucleus of the research programme. The simulator-based HAMMLAB laboratory constitutes a focal point of the Project’s man-machine system research where the human factors programme relies upon HAMMLAB for performing the experimental programme while new Methods and Technology are developed, tested, evaluated and demonstrated in a realistic environment.

6.1. HAMMLAB - a basis for the research programme

The first HAMMLAB became operational in 1984, but upgrades and extensions have taken place several times and the current version - HAMMLAB 96 - now represent the

![Diagram of HAMMLAB's main structure and functions](image)

Figure 1. The main structure and functions of HAMMLAB 96.
forefront of current control room technology with a unified MMI, an advanced intelligent alarm system, several types of overview displays, safety parameter display system etc. The main structure and functions of HAMMLAB 96 are illustrated in Figure 1. The facility has three major functions:

- **process operation** - in the control room - where the NORS process is simulated and operators (test subjects) are monitoring and controlling the process in normal and disturbed plant conditions,

- **conduct experiment** - in the experimenters gallery - where the experimenters set up, monitors and control the experiments, and where a database is collected during experiments consisting of human performance measurements as well data from the control room operation and the process itself,

- **evaluate experiment** - in the experimenters gallery - where the experimenters analyse results from the experiments by means of various techniques and statistical packages.

### 6.2. HAMMLAB 2000

To achieve the goals of increased number of experiments and useful output from the facility it has been decided to expand the facility in terms of flexibility and functionality. The major requirement is to provide a new modern simulator of western type PWR and BWR - or both - with a broad operational domain, including simulation into accident states, and which is developed using modern software tools for enhancements of maintainability and flexibility. Such a western simulator will hopefully also give access to more plant operators to take part in experiments in HAMMLAB.

The second major requirement is to provide space enough such that an advanced control room including large screen displays can be built. It is strongly believed that such control room settings will be the environment for future control room operators.

The HAMMLAB 96 facility will of course be kept such that the research programme can continue at full speed while HAMMLAB 2000 is being built. It is believed that the new facility - HAMMLAB 2000 - will increase the possibilities for closer co-operation with member organisations and transfer of results from the Project to direct applications.

### 7. Summary

Backfitting of nuclear power plant control rooms is a continuing process, introducing computer-based solutions for surveillance and control as well as for improving the
human-computer interface. At the same time designs for tomorrow's reactors are developed, characterised by fully digital instrumentation and control systems, and advanced, computer-based control rooms. Research and development efforts are needed to ensure that the new technology gives the expected improvements in operational safety and efficiency.

The research programme at the Halden Project addresses the research needs of the nuclear industry in connection with introduction of digital I&C systems in NPPs. The programme provides information supporting design and licensing of upgraded, computer-based control room systems, and demonstrates the benefits of such systems through validation experiments in the simulator-based experimental control room facility at Halden.

At the Halden Project an internationally sponsored research programme is carried out which addresses these research issues. The Halden Man-Machine Laboratory represents a unique test-bed for investigating new, computer-based solutions for nuclear power plant control rooms. The research programme draws upon competence built up through more than 25 years work in the field of computer-based operator support and digital control room solutions, and the close contact with licensing authorities, utilities and reactor vendors in the 19 countries participating in the Halden Project ensures that the work is addressing the real research needs of the nuclear industry.

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Regulatory Perspectives on NPP Simulator Applications

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1. Background

After the last specialist meeting on simulators and plant analyzers it was foreseen that the majority of the applications fell into the following groups:

- Simulations for operator training
- Evaluation, design and testing of procedures
- Generation of data for PSA
- Studies of human behaviour and reliability
- Operator assistance
- Safety analyses

The simulator applications on which the regulatory bodies have traditionally mostly focused are operator training, evaluation of procedures and safety analyses. The other areas are more often followed at a distance.

The operator training has been a significant issue. The power of the control room replicas as a training tool was recognized, and attention from the authorities has mostly been focused on the use of such equipment for operator training. In some countries regulations for such use of simulators were issued and the equipment was also used, for instance, for examinations of operators. The general judgement from the authorities is that simulators used for operator training purposes should be as close replicas of the control rooms as possible and that the simulation should be sufficient for the general features of plant behaviour to be recognized. There has been a significant reluctance to accept training on accident sequences which have a high uncertainty. For such occurrences other training methods are recommended.

Inspections of procedures for preparedness and emergency operating procedures in Sweden have resulted in a recommendation from SKI to use existing full scope plant simulators as a tool for validating such procedures. It has been identified that valuable information were obtained, in particular of the work load on the plant personnel under accident conditions.
The third area of special interest for the safety authority is the safety analyses using advanced plant analyzers. New safety analysis methods based on best estimate simulations of progressions of accidents have been developed or are under development. Such methods will be used much more in the future. Normally large numbers of sensitivity analyses will have to be performed in order to evaluate the accident sequences and the uncertainty associated with them, and also validation of the methodologies by comparison between calculations and experiments and plant data is needed. The recent rapid development of computational resources have made such methods feasible. As a consequence efforts will be devoted to development of, for instance, Graphical User Interfaces in order to facilitate communication with the code users. In fact, this is one of the original ideas behind the plant analyzers.

The application of simulators to PSA-studies has so far been less successful. An old objective was the attempt to model human behaviour and interactions in the PSA-studies. Results of integrated safety analyses indicate that this objective is not very likely to be reached, and that human behaviour has to be treated differently.

2. Future safety demands

Large modernization and back fitting projects are planned or being carried out in different countries. The particular objectives of such programs may vary. The background for safety considerations is to upgrade nuclear power plants built to earlier standards, to standards required for newer reactors.

An important force behind future safety considerations is the increased conscience of the public about risks and safety, and nuclear energy is for the time being considered by the public as a safety problem. Other such forces are the need of the plant owner to protect the large investment, the increased knowledge through experience, analyses, research and development, and perhaps also revised safety criteria and international conventions and work with modern safety concepts.

Very important aspects of the communication between the representatives of the nuclear area and the public are the application of openness and the establishment of confidence. The competence of the nuclear representatives is in this context very important. The experience has shown that confidence among the public is eroding quickly if a plant has to be shut down or operate on reduced power for reasons which are not fully understood. Unplanned interruptions are not welcome, and a shutdown of a plant because of an accident or a safety problem is a nightmare. Several such situations have occurred in Sweden.

Another very important aspect for the acceptance of nuclear power is that our technical and organizational measures to achieve safety are robust and simple so that they can easily can be explained.

The safety analyses of the reactors will in the future need to encompass a much larger spectrum of hypothetical incidents.
3. Role of plant analyzers in fulfilling the new demands

The origin of one of the Swedish modernization programs was identified defects, and the reactor was out of operation for more than 3 years. In the report from the utility to SKI it was noted that the problems would have been detected earlier if they had been detected and analysed more carefully in the past. Another large recent issue in Sweden was the wrongly adjusted setpoints of the secondary safety valves. This was actually a 54-fold common cause failure. It is probable that the failure would have been noted earlier if appropriate analyses of previous incidents had been performed.

The position of SKI in this respect is, in principle, that all transients, incidents, and anomalous behaviour which occur in the reactors should be carefully analysed using modern analytical tools such as advanced plant analyzers, if needed. Emphasis will also be laid on answering so-called “What if?”-questions. The main objective is to verify that no safety limits have been exceeded, in order to be able to make a complete safety assessment of the occurrence, filling in the “gaps” using advanced methodology and in certain cases to be able to assess the analytical tool used. An important safety aspect is, by a broadly based analysis, be able to assess how the safety systems would behave during accidents. The assessment of how systems behave under mild challenges, is perhaps the best source of information in this respect.

One observation is also that it is often the small, trivial-looking, incidents that have the potential to develop into a major threat or accidents. It is therefore a growing concern that these potential accident initiators can be identified and remedied.

This approach would require that data from occurrences are recorded and stored in such a way and for so long time as needed to carry out the analyses. The requirement to collect and store data from occurrences, and also to analyze safety impacts, already exists today. The major problems have been that only a few parameters have been collected, that insufficient knowledge about the dynamic response of the instrumentation has been realized by analysts and plant personnel, and that advanced tools, like plant analyzers, have not been used in the analyses. SKI would like to see major improvements in terms of completeness of collected data to fulfill the demand, and depth of analyses. SKI cooperates with the industry in a project for comparisons between plant transients and analytical results.

The Swedish nuclear power industry conducts large projects with the goal to revisit the design bases for their reactors. New FSARs will be produced and the basic assumptions will be checked. New analyses, using modern methodology, will be applied to assess the safety significance of some of the limiting transients. New findings have indicated that the safety analyses for some cases should be carried out much longer time so it also covers safe permanent cooling of the plant. The role of plant analyzers as validated best estimate methodologies, is in this respect important.
4. Closing remarks

During the last meeting a statement of one of the TMI-2 operators were quoted: "If an operator should be part in a severe accident, he should at least have a chance to learn about it in beforehand. An operator should never be put in a situation that has not been analyzed." The objectives indicated by these words are still not completely reached but we are under way.

By the recent development of fast and inexpensive computer resources it is now possible to perform numerous analyses in great depth and detail. The plant analyzers should be used more extensively to improve our safety assessments. In particular it is believed that regular in-depth analyses of plant occurrences is a very powerful tool for identification of problems in safety systems and accident initiators.
Summary and conclusions
of the second OECD specialist meeting

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- When? 8-10 June 1993
- Where? Halden Reactor Project
- Organized by SESAM, as a follow-up of a
  recommendation of the Rome 91 meeting
- Proceedings: NEA/CSNI/R(93)9
- Conclusions: NEA/CSNI/R(94)13
- CAM and PWG4 discussed the need for new meeting
- Progress was noted in this area since 1993
- Still conflicting views on use of operator aids and their capabilities
- Second meeting endorsed by CSNI
- Scope could be much the same as SAMOA 1

Scope

- Operator aids for accident management (AM)
- In operation or soon to be
- AM for beyond design basis accidents
- Including tools which might be extended from the design basis range to the SA area
- Including relevant simulation tools for operator training
- Focus on results and applications
Objectives

- Review recent developments
- Discuss feedback from:
  - emergency drills
  - training sessions
- Discuss the validation of operator aids
- Identify areas of international
  - agreement
  - disagreement
  - open issues

SAMOA-2

- When? 8-10 September 1997
- Where? EDF/SEPTEN, Lyon
- 33 experts from 10 countries, HRP, CEC and RF attended
- 20 papers were presented
- 2 demonstrations
- Proceedings and Conclusions to be published
General Conclusions

- Development and implementation of operator aids for AM is in progress
- Tools presented are essentially the same as at SAMOA-1, but
  - enhanced to a considerable degree
  - extensively validated
  - successfully used during drills
- Implementation slower than expected
- Training for SAM is considered beneficial
- There is still debate on the need for advanced computerized tools in SAM

Operator Training

- Knowledge-oriented or skill-oriented?
  - choice is still open issue
  - simulators useful in both cases
  - skill-based training probably requires faithful real-time simulation
- If OAs are implemented, they should be tested during training sessions
- Regulators do not plan to consider SAM skills to be part of formal operator licensing
- Training for SA is still in infancy; there is room for international collaboration
Operator Aids

- OAs have to be integrated with SAMGs
- Still no consensus on the usefulness of computerized tools for SAM at the plant
- If computerized tools are to be used for SAM at the plant, then provide same MMI as for normal operation to make operators familiar with the tools

Areas for Improvement

- Information validation (including signal validation) is important but not yet mature
  - fuzzy logic and ANNs are being tested; they look promising but qualification and formal proof of reliability not solved
- Good communications and understanding: be cautious if different tools are used by different teams at different places
  - emergency organization must cover resolution of conflicting conclusions
  - if consistent tools are used, communication is facilitated
- Use of Internet to improve sharing of information?
- Unanimous agreement that international collaboration should be increased:
  - development of computer-based OAs
  - OAs applicability to operator training
  - training programmes for SAM
  - most appropriate simulators for SAM training
ALICES: an Advanced Object-Oriented Software Workshop for Simulators

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ABSTRACT

Reducing simulator development costs while improving model quality, user-friendliness and teaching capabilities, is a major target for many years in the simulation industry. It has led to the development of specific software tools which have been improved progressively following the new features and capabilities offered by the software industry.

Unlike most of these software tools, ALICES (which is a french acronym for "Interactive Software Workshop for the Design of Simulators") is not an upgrade of a previous generation of tools, like putting a graphical front-end to a classical code generator, but a really new development. Its design specification is based on previous experience with different tools as well as on new capabilities of software technology, mainly in Object-Oriented Design. This allowed us to make a real technological "jump" in the simulation industry, beyond the constraints of some traditional approaches.

The main objectives behind the development of ALICES were the following:

- Minimizing the simulator development time and costs: a simulator development consists mainly in developing software. One way to reduce costs is to facilitate reuse of existing software by developing standard components, and by defining interface standards.

- Insuring that the produced simulator can be maintained and updated at a minimal cost: a simulator must evolve along with the simulated process, and it is then necessary to update periodically the simulator. The cost of an adequate maintenance is highly dependent of the quality of the software workshop.

- Covering the whole simulator development process:

  - from the data package to the acceptance tests and for maintenance and upgrade activities;

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- with the whole development team, even if it is dispatched at different working sites,
- respecting the Quality Assurance rules and procedures (CORYS T.E.S.S. and TRACTEBEL are ISO-9001 certified).

The development of ALICES was also done to comply with the following two main constraints:

- Building an open system: ALICES will evolve and need to interact with foreign systems; new tools will easily be added.

- Building a modular system: ALICES will also be used for simulator upgrade and update. It will be possible to integrate ALICES progressively into an existing simulator maintenance and evolution process.
1. Introduction

With the rapid power increase of computers and their ever decreasing price, the industrial simulation world is facing two major challenges.

On the one hand, progress is expected in model quality, in user-friendliness and in teaching ability. With each step of progress in one of these fields, simulators become even more interesting educational solutions to new categories of potential customers.

On the other hand, the building of better simulators requires various types of expertise, especially in computer and software technology, in process simulation, in ergonomy and in graphic drawing. As a result, highly skilled personnel is needed and simulator costs are nowadays strongly dependent of these miscellaneous software issues. As an example, software development represents around 80% of the development costs of a middle size training simulator in the electrical industry.

For many years, special software tools have been designed for easing the development of simulators, and thus reducing the development time. They achieved their goal by facilitating a few time consuming tasks of the development process.

Though this was already a big progress, a lot of work remained to be done, especially in integrating the various software components and testing them properly. Following generations of simulation software workshops tried to improve the solution to these problems.

Unlike these tools, ALICES is not an upgrade of previous tools but has been developed from scratch as a coherent, integrated and appropriate answer to the building of modern simulation systems. It is a new generation engineering software workshop that includes all what is necessary to build and maintain various types of simulators.

Indeed, apart from offering sophisticated and modular modelling tools, ALICES takes care of important simulator development aspects such as management of input data, product testing, progressive and modular integration, documentation, project development and co-ordination.

ALICES has been developed by a Consortium of European companies, among which CORYS T.E.S.S. (France) and TRACTEBEL (Belgium), two companies which are world leaders in the simulation business. ALICES has been awarded the European EUREKA label.
In this paper we first expose the way ALICES works, its objectives and constraints; we then present different important aspects in a simulator development and how there were treated. Some technical information is then provided.

2. ALICES objectives and constraints

The main objectives behind the development of ALICES were the following:

Minimising the simulator development time and costs: as already mentioned, a simulator development consists mainly in developing software. One way to reduce costs is to use a generic simulator core coupled with software components describing the simulated real world objects. Moreover, since many of these components can be designed with generality in mind, they can be saved into general standard component libraries and be ready to be used in many projects.

Insuring that the produced simulator can be maintained and updated at a minimal cost: a simulator must evolve along with the simulated process, and it is then necessary to update the simulator as the process is upgraded. The cost of an adequate maintenance is highly dependent of the quality of the software workshop.

The development of ALICES was also done to comply with the following two main constraints:

Building an open system: ALICES will evolve and need to interact with foreign systems: new tools have to be added easily. Moreover, the ALICES architecture allows it to be linked with already existing object components that have not been specifically developed for ALICES.

Building a modular system: ALICES will also be used for simulator upgrade and update. ALICES structure is such that affective recovering of other simulator software pieces is possible. This is the key to allow a smooth transition from an old generation simulation system to ALICES.
3. Important aspects of ALICES

To fulfil the requirements given in the preceding section, special attention has been given to the following aspects, which are essential to an appropriate simulator development and maintenance. We present here the solution that has been chosen within the software workshop ALICES.

Input data

It is very important to trace the original process data, to know which value is being used, what kind of previous calculations, hypotheses or tuning have been performed on the data. ALICES offers in this area an integrated data base facility for keeping and handling the original process data through integrated calculation parameters sheets. These are the link between the data used by the simulator and the original input process data. Thus, it is very easy to know how the value of each parameter is computed.

It is also possible to identify which data have been tuned to match the simulated process with the real one.

With the help of the built-in documentation facilities, it is thereby easy to keep trace of all the data and the tunings used. Such a knowledge is essential for a proper simulator maintenance and evolution.

Tests

Adequate test facilities are primordial to guarantee an efficient simulator development. The sooner and the deeper a model is tested, the lesser it contains potential problems that could appear only at the simulator development final stage, resulting in time and cost expensive debugging sessions. It is also important to know what has been tested. ALICES provides in this field various tests facilities, including the possibility of writing and replaying test scenarios.

Tests can be performed as soon as a group of objects is assembled. It is not necessary to waste time in compilations, it is just necessary to define the model that is going to be tested, which is also necessary for assuring an easy trace of the tests performed.
Documentation

Documentation is an essential aspect when building and maintaining a simulator. ALICES offers all the facilities necessary for creating, printing and consulting on-line documentation. Whatever is developed with ALICES can be easily and interactively documented.

Project development

A proper organisation is an essential key to succeed in a simulator development. ALICES provides project development tools, allowing the configuration and follow-up of various types of projects, control on the rights of use and general organisation of the software development to be performed. These tools have been devised to cope with a proper development methodology. For example, it is easy to allocate to each user the tools necessary for his work, so that starters with the workshop do not get overloaded will all the existing possibilities.

Co-ordination

In any software project involving more than one developer, co-ordination is a critical item. Much time can be wasted whenever people working together are not communicating properly.

This problem has generated a special attention to the development of ALICES by helping developers to interface properly their respective applications. The use of gigantic databases is avoided, and most of the interface labels between systems built by different people can be generated automatically, with the help of the process input data. This greatly reduces potentially serious co-ordination mistakes.

4. How does ALICES work?

Let us consider first the efficient three level approach used to analyse and design a complex process in the industrial world. Such a process can be described in terms of basic objects strongly interacting together inside subsystems, while the subsystem assembly forms the entire system.

The main point with this approach is that a single object or a subsystem can be described, analysed and simulated much more easily that the whole system. Besides, a direct advantage of this approach is to ease the system upgradings, since an upgrading concerns often only a small subset of one or a few well-identified subsystems, the other ones being not affected. ALICES organisation features naturally this three level approach (Fig. 1).
Figure 1. The architecture of a simulator development with ALICES.
Level 1 is used to create the basic objects of the simulation. The dynamics of these objects is described by discrete time transitions from one object state to another or by ordinary differential equations (ODEs). Moreover, each object can be made fairly general by incorporating parameters in its definition. This allows the building of general object libraries reusable within different simulation projects.

Two major features have been incorporated in ALICES to simplify the modeller's task.

Firstly, each object needs to communicate with other simulated objects. In ALICES, this is achieved in two ways, depending on the fact that the objects that need to communicate belongs or not to the same subsystem. Inside a subsystem, the objects are usually tightly coupled. This is expressed by graphical connections via special input/output points called “connectetypes”. Each connectetype is defined to exchange a given information in a given direction (from or to the object) and the exchanged information is directly available within the object. Moreover, for a connection to be valid, the exchanged data through two connectetypes must match in type (what is exchanged ?) and in direction (from an object to another object). This strongly reduces the communicating interface problems between two objects that have been created by two different persons.

For objects belonging to different subsystems, another communication facility is available. Each object has the ability to publish every quantity it computes under a public name. Through this public name, other objects can access the published quantity. Here again, to avoid mismatches, each public name is checked to be generated by only one object.

Secondly, the other major feature that simplifies the modeller's work is the possibility to express the ODEs describing an object state evolution in terms of secondary variables instead of the object state variables. All that has to be done is to express in a very natural way the relation existing between each secondary variable and the object state variables it is function of. Moreover, the Jacobian matrix of the secondary variables relative to the state variables is computed automatically. This enables the object dynamics ODEs writing to be simpler compared to what it would be if it was expressed as state variable functions. As a result, the objects are created faster and are less error prone.

Once an object exists, its behaviour can be tested by creating a simple system made of this object connected to passive or active boundary objects. After it has succeeded its unitary tests, the object is ready to be assembled with other checked objects into subsystems. The object unitary test and subsystem realisations feature ALICES Level 2.
At level 2, the objects created at level 1 are graphically dragged onto a virtual working document and connected together via their connectypes. The object parameter values are also set. Here again, a special care has been taken to simplify the modelling work and to avoid errors since the parameter values can be read or computed from external databases. Level 2 is also responsible for giving a public name to quantities that need to be exported to or imported from other subsystems.

The subsystems are assembled together and run at level 3. The subsystem set can be the whole system; however, for test purposes, it can consist of only a subset of these subsystems.

Level 3 gives the user the ability to follow the evolution, textually or graphically, of every quantity used in the simulation. The users can also modify on the fly these quantities.

Level 3 has also been designed to allow external code integration with the simulator. It should be noted that compilation occurs only at level 1, in order to produce the software components we talk about in section 2. These components need then just to be dynamically linked with the simulator core at level 3 to get an executable simulator. That feature greatly speeds up all simulator modifications.

5. Technical background

ALICES was developed in the industry standard C++ language. The object oriented development method, originating from G. Booch’s work, was chosen to guarantee coherence during the building and the successive evolution phases of the workshop. ALICES runs under the Solaris operation system on Sun SparcStation. The subsystems can be shared over different processors and/or over different workstations. Special care has been given to insure that ALICES is portable to other UNIX environments and to Windows NT.
6. Conclusions

ALICES has been designed in order to satisfy the following important points of view:

The point of view of the project: time and costs of simulator development and maintenance need to be minimised.

The point of view of the designer: modularity and open systems are required to insure easy interface and integration with already existing systems.

The point of view of the user: flexibility is essential, in order to adapt the software workshop to the size of the development, user friendliness and coherent environment are a must in order to facilitate the use of the software workshop.

As an open system, ALICES can also be used for building other types of simulators, such as design simulators.

The software workshop ALICES is currently used to develop the multifunctional simulators at TRACTEBEL.
CISO : Charter of Integration for Simulator Openness


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1. Introduction

CISO is a set of rules and guidelines for simulator interoperability and openness.

1.1 Context

NPP simulators at EDF are mainly used for training and studies. Full scope simulators are used to teach the operational procedures, whereas smaller scope simulators such as SIPACT are used to develop the understanding of the underlying physics, thanks to a CATHARE based NSSS model. Simulators are also used for studies and in the future they will be used for engineering, in a consistent relationship with CAD databases.

Whatever the usage is, a simulator comes out as a simulation machine connected to some "pre-processors" (instructor station, control panels, CAD) and "post-processors" (didactic stations, graphics tools, validation tools).

More specifically, the new generation of PWR full-scale simulators at EDF, also named "projet SIROCCO", shows, among others, the following distinctive features:

- size: ten full-scale simulators are concerned (under four plant series),
- homogeneity: thanks to EDF's standardized plant series effect, only four series must be considered,
- numerical models: a derivative from the french thermalhydraulics code CATHARE 1 is used for real-time simulation of the NSSS (see B. Pentori contribution), and a real-time common version of CATHARE 2 is under development for future integration,
- sub-systems openness: (computer) subsystems that do not need to be specific for one series are replicated and connected to any type of simulator. Only the process simulation subsystem and the hardware replica of the control room are specific to a plant series. These two subsystems are built as pairs, each pair coming from the same vendor. The other subsystems (the main operating desk, the training session staticitical analyser, the software replica of the control room, the data acquisition and safety panel) need not come from the same vendor as the simulation subsystem.
- call for bids openness: bids are asked on a computer subsystem basis

In this context, it comes as a necessity to define a charter for all the subsystems to integrate and to evolve easily. CISO is the technical counterpart for those requirements.

1.2 Objectives

The objectives are the following:
• build an integration framework for NPP simulators simultaneous developments, which will be used as a requirement for the next generation of full-scale NPP simulators,

• improve components and tools reuse among projects,

• simplify maintenance operations.

As EDF requires external vendors to build the simulators, the rationale for this prospect is:

• focus on interfaces between simulators subsystems and teams,

• develop a common "language" for describing and sharing simulation objects,

• do not interfere with systems implementation, in order to respect everyone skills and duties.

1.3 Scope

After careful examination, the real issues were identified to be mostly a matter of interfaces: how easily can teams interact, how can one exchange simulation modules to and from simulator development environments, what is the optimum in inter-systems communication with respect to design complexity and communication throughput.

![Diagram](image)

Figure 1: CISO addresses three main topics: 1/ homogeneity: what are the rules that allow the development teams and the users to interact with any simulator project using the same "language"; 2/ code integration: how can the NSSS code and its future version be integrated and maintained in the simulators at the lowest cost; 3/ interoperability: what must be stated for subsystems from different sources to communicate one with each other, regardless of supplier's specific methods.

This is why CISO comes out as a set of rules and interfaces. Rules are intended to develop a common understanding for all the important concepts such as: simulated components, malfunctions, variables, types and units, simulation modules, initial states, time management, etc. Once the rules are given, it is possible to define and develop unambiguous interfaces to exchange data, be it for configuration or for operation purposes.
2. CISO in brief

2.1 The Static Data Model

CISO starts with the definition of a common language for describing static data. The language scope and format is inspired from SIPA, from EDF engineering rules as well as from EDF CAD standards.

Static data is the specification for the simulated installation (e.g. topology, variable type), as opposed to dynamic data (e.g. variable value). Static data describes: the geographical extent for each simulation module, malfunctions, local commands, model controls, variables types and names, references towards datapackage.

For this language CISO first defines a semantics, by giving a definition to all the useful concepts, then a syntax is defined, giving a uniform text file representation for the static data.

Figure 3: The static data model is in some sense a repository from which different view can be extracted. It only needs to exist outside of the software development environments.
It is important to mention that all the entity types are related through the model in a consistent manner. For instance, from a reference to the datapackage, one can reach a component which owns variable names whose values are computed by some simulation module, etc.

The static data abstract model is given through the definition of entity types, relationships, and naming conventions. There is a text for explanation and an object model for reference and global coherence. The subsequent static data model is just an interface to the SDE, not an implementation requirement for it.

2.2 Simulation modules description

Based on the naming conventions for variables and constants, CISO defines a format for simulation modules (or ciron) description and exchange. This format has been inspired from SIPA modules architecture, distributed object models and EDF engineering rules.

The objective is to be able to exchange simulation modules in a parallel yet deterministic environment. For this purpose, CISO defines a typology for modules (based on the functional role, such as "thermalhydraulics" or "control"), so as to get homogeneous geographical extents and interfaces. Rules are also given for inter-module communication, so that deterministic parallelism can be achieved.

The "ciron" shows an interface data specification (in, out, constant) based on the static data model and the computational part of the code. The ciron does not show the actual implementation for data handling and exchange, nor the interface with the real-time infrastructure.

Basically, the exchange syntax contains the lines of code for the numerical models, leaving ignored those that are specific to the real-time interaction with the simulator infrastructure. This allows for an abstraction / implementation systematic procedure for exchanging ciron, and specifically those derived from CATHARE.

```
uses
    connection GCT010VV field OPEN :: variable X[1] / IN
    connection GCT502XU field 204 :: variable X[2] / IN
    connection GCT202AA :: variable Y[1] / OUT

private
    int X[10], Y[10];
    service INITIALISE
        Y[1] = 0;
    service RUN
```

Figure 4: A trivial example for a ciron which describes a code performing an AND operation, initially written in C language. The ciron refers to the connexion points (here for instance GCT010VV) which contain typed fields. Fields are oriented and associated to the internal variables of the module. The ciron shows the computations performed by the module under several circumstances such as initialisation, normal operation, saving an initial state, etc. The ciron never shows the actual implementation of the data exchange or interaction with the real-time infrastructure, which remain free.

2.3 Communication between subsystems

The objective is to make the communication process independent from the parties' specific techniques, to make the interface configuration phases easier, to devise a simple highest-level
communication protocols to reduce interactions, and to use a standard, dependable and lasting communication system between simulator and clients.

For this purpose, CISO fixes the network infrastructure up to the application level. The applications are required to connect to the network through a most simple API (asynchronous get & put). The interfaces are configured using standard messages description syntax and compilation techniques. Finally, rules are given to simplify synchronism constraints (initial states management, time management, state transitions, etc.) and to reduce throughput between subsystems. These high-level application protocol rules aim at maximising the independence of the parties' technical choices.

Figure 5: CISO makes mandatory the usage of a standard communication subsystem, based on widespread technology. The interface between the application and the network is provided through a limited set of entry points, on the top of which suppliers can implement their own techniques. The messages description syntax is shared between parties, so that the configuration of the exchanges can be performed coherently. The methods for message coding are also shared.

3. Summary

In order to improve NPP simulators components homogeneity and reuse among projects, EDF has defined a Charter of Integration for Simulator Openness. CISO defines a static data model which must be used for any reference to the simulated installation, exchange procedures for simulation modules in a parallel environment and a standard basis for compatible communications between the simulator subsystems. CISO concentrates on interfaces and leaves open the individual choices for implementation.

The prototyping phase is now completed, CISO will be put into practice for the next generation of EDF full-scale simulators, the guidelines must now enrich from experience.
APROS - A MULTIFUNCTIONAL MODELLING ENVIRONMENT

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Abstract

The Advanced Process Simulation (APROS) environment has after more than a decade of dedicated product development and intense commercial use reached a level of maturity that is difficult to find with regard to similar products. One of the basic ideas behind this software tool is its multifunctional concept. The concept requires that the tool is suitable for modelling and simulation of the dynamics of a process plant during all phases of its life-span from pre-design to training and model supported operation and control. The implementation of this concept had a significant impact on the software structure. Several, sometimes contradictory requirements had to be encompassed. It should be suitable both for small simple models and full scope simulators. It should facilitate time-steps from milliseconds to minutes, for the same models, just depending on the scope of study. It should combine several modelling paradigms such as continuous, discrete, mechanistic and empirical. The intrinsic model building blocks should be comprehensively verified, but users' model equations should be accepted, as well. It should be easy to connect to external models or hardware, and to use both in master or slave mode. It should be easy to study and modify the internals of the models, their structures and parameters, but it should also be possible to disclose all delicate model information from unwanted access. The calculation should be optimised for current computer hardware, but the model specifications should be easily transportable to new platforms. And finally, it should be suited both for researchers, engineers and plant operators. How did we succeed? We had 20 years of comprehensive thermal hydraulic modelling tradition before starting the project. We had the key experts with the key knowledge. We dedicated more than 100 man years of efforts for the new software developments. Presently, we have a superb team maintaining and improving the software, complemented with new enthusiastic members.
1. Introduction

The development of computer hardware and software during the last ten years has considerably changed the status of traditional nuclear safety analysis software. Affordable computing power on the safety analysts table by far exceeds the possibilities offered to him/her ten years ago. At the same time, the features of everyday office software tend to set standards to the way input data and calculated results are managed and presented.

Current requirements on nuclear safety analysis code are not any more limited to correctly predicting the plant primary circuit behaviour in certain anticipated operating conditions. In fact, a large number of possible severe transients originate from minor malfunctions in automation or electrical power supply systems. These systems affect the plant behaviour significantly through the secondary systems, which also have to be included in the model.

Accurate and fast enough models can be used both for plant design and personnel training. Design specification data including the actually used decision criteria is included in the information collected during the whole plant life cycle to be available for the planning of plant modifications and for the training of plant personnel. Such data is by preference accessed from the same graphical diagrams which are used for the model specification. The software can help to transfer design knowledge to the users of the plant. When changes and updates to the plant design have to be made, much less re-engineering work is needed. Also all changes to the design can be tested in advance on the simulator, before application to the real plant.

2. Physical models

The physical models must be uniform in the sense they can be interconnected to each other using a certain set of simple rules. Wide application range is important as well. Another feature is the robustness of the solution algorithm, especially during abrupt changes of the actual model structure e.g. when a large leakage is introduced. The basic entities should be as few as possible. The nodes and branches of a conventional flow network can actually describe flow of any material, if the parameter and correlation system can be adjusted accordingly. Thus the basic network model is applicable for e.g. the flue gas network as well as for the primary circuit.

Thermal hydraulic models range from homogeneous to five and six equation models, depending on the needs of current analysis task. Single phase flow models may be needed for fast simulation of the numerous auxiliary systems. Reactor models with two energy groups should range from 1-D to full 3-D models, all with user selectable amount of flow channels, including a set of hot channels with all necessary adjustment parameters. The reactivity level and axial power distribution should be tuneable as well as the feedback effects, which should have additional conservativeness factors to be used according to safety analysis needs [1]. A modern analysis tool must have proper facilities
to describe present day automation systems. A possibility to actually use the manufacturer
specific components or software is an advantage. OPC technology is used for connecting
APROS to modern control system software. The electrical system to solve the electrical
power parameters in the plant network is often useful considering the amount of
transients actually initiated by a fault in the network.

Validation of the code is essential. The standard model of validation is still valid:
physical model verification, comparison with separate effects tests, studies of a full
plant behaviour. A multifunctional code brings along a much larger set of validation
cases, if its wide application area as a design tool can be exploited properly. Always ask
for the validity range of a thermal hydraulic simulation software. Is it suited for real time
operation in a full scope simulator on affordable hardware?

3. User interfaces

The user interface is informative, self-explanatory, easy-to-use and at the same time it
supports the needs of standard procedures in building the models and using them. The
software is capable of telling the user about configuration errors he/she has made or is
about to make. After completing a model for a process component or a sub-process the
user can easily test its functionality. In the course of building the model the user
interface supports different visualisation modes, which are used to show and highlight
certain model parameters used around the model in order to help the user to detect
possible inconsistencies in the model. During the simulation the interesting parameters
are available in the same manner.

4. Layered architecture

The multifunctional simulation software has a multiple layer architecture. It consists of a
kernel part, which includes the database, solvers, basic physical models, and
communications interfaces e.g. to the user interface. This layer is closed to the users of
the software because of validation consistency. The next layer is the configuration
layer, where higher level process components are defined making use of kernel level
elementary components. The following level is the application model layer where
processes or sub-process are specified by the process components. It is also possible for
the user to add on own models as DLL subroutines coded in C or FORTRAN.

The database system is evidently the basis of all the functionality of the simulation
software. It is capable of taking care of interconnections between models, cutting
models apart, copying and renaming them and at the same time storing not only process
related data but also the relevant data for the graphical user interface. This data is stored
in the same database as the model data in order to provide strict correspondence
between the actual process model and the picture, which in turn is used as part of the
model documentation.
The possibility to exploit the database to define which models are actually included in the simulation also facilitates the alternate use of fast simpler models and slow accurate models in the same simulation system. The database has facilities for referencing component, process, plant or project related data, which describes sources of input data, design and modelling criteria and also sources of validation data, validation results and the statement of validity range of the model.

It is required that the database also has a system to define a set of access rights for the project personnel indicating the components and subsystems, which are open for modifications by a certain person. This feature facilitates continuous development of a model during the plant life cycle.

5. Advanced applications

The multifunctional simulation software system can find many advanced applications along with the development of affordable computing power. APROS runs already on a laptop computer. A full scope training simulator is possible to run on a single powerful workstation. APROS has been used both for full scope and compact simulators. Several types of PWR and BWR plants have been modelled. Modelling of VVER plants has been done in co-operation with our development partner Fortum Engineering Oy [2]. By experience, APROS makes it possible for the end user to construct his own training simulator, thus saving a lot of money. APROS is really proven technology.

The really challenging applications include tracking mode and prediction simulators. The former is used to estimate the state of the plant even through accident sequences in order to provide up-to-date initial conditions to a prediction simulator used for faster than real time studies of possible choices in operational procedures. APROS will be prepared for modelling of severe accidents.

6. Upward compatibility

The standard tools, computer operating systems and graphical display software will develop at an unknown but fast pace. To cope with the changes it is essential that the basic kernel and the application dependent part of the code are developed in a strictly modular way, taking care of well defined interfaces between the modules. It is easy to transport the application model specifications between different computers having different operating systems or even different bit or byte patterns used to reflect the values of data. The whole model or a part of it can be converted to an APROS Specification Language (ASL) file, which can be used for storage of sub-models or for transportation of models between different computer platforms [3]. Accordingly the modelling work survives to new computer generations.
APROS is in use on four continents. The numerous models already configured during the years by our customers or ourselves represent a major investment for the future. Re-engineering and re-modelling can be avoided. We are committed to continuously develop and improve the software and are open for new partners in our network of distributors, added value developers and end users.

References


THE TREND TOWARDS WINDOWS NT FOR USE IN SIMULATION

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ABSTRACT

The move from Encore and UNIX computers to PC-based and DEC Alpha Windows NT platforms in the training simulator industry has been swift. Nearly every nuclear plant simulator platform upgrade in the U.S. over the last two years has been to a PC system, and the first contract for a new full scope PC-based nuclear plant simulator has recently been awarded. The advantages of this approach are obvious, from the ease of maintenance and future upgrades to the cost savings from decreased electricity requirements. However, there are still lingering doubts among some in the industry about whether or not PC’s and Windows NT are ready for large scale real time applications, such as running a training simulator.

Much of the software and hardware required to connect a PC to simulator I/O and run in real time is based on standard third party applications and off the shelf hardware. This is in contrast to the custom made, mission specific software and hardware traditionally utilized in the training simulator industry. What limitations are incurred by using standard software and hardware? Have PC and Windows NT simulators achieved reliability levels as good as the platforms that they have replaced? Have the advantages of running on PC’s turned out to be real or are they hype? What technical problems still exist? There is now enough of an operating history to address these questions.

THE HISTORY OF WINDOWS NT REAL TIME SIMULATION

Several events in the summer of 1994 led to the development of real time Windows NT simulation capabilities. Windows NT Version 3.51 was introduced, which provided a stable, industrial strength, multiprocessor-capable operating system for Intel-based PC’s. Intel Pentium processors achieved a level of performance previously seen only in expensive workstation and mainframe computers, and became available in multiprocessor configurations. To complete the picture, Fletcher Gibbs and Jim Knight of the Tennessee Valley Authority developed and released to the public domain a UNIX-based real time simulator executive called OpenSim.

RNI Technologies, then known as Ryan Nuclear, Inc., in Columbia, Maryland obtained a copy of the OpenSim code and
set to work porting it to run under Windows NT. Vermont Yankee provided funding for the effort, and agreed to become the first training simulator to run on a PC-based platform using the Windows NT version of OpenSim, designated OpenSim NT by RNI Technologies. The Vermont Yankee training simulator was running on the new platform in early 1995.

The Monticello training simulator was next to port to Windows NT (including the first Windows NT instructor station), followed by Cooper and Three Mile Island. Following the successful completion of these upgrades, other vendors began to introduce Windows NT-based simulator executives. In November, 1996, the first contract for a new full scope simulator based upon an Intel/Windows NT platform was awarded. The Swiss utility Kernkraftwerk Gösgen (KKG) contracted STN-Atlas (Germany) to develop a full scope simulator for the Gösgen nuclear plant using the RNI Technologies OpenSim NT simulator executive and OpenSim IS instructor station.

USE OF STANDARD VERSUS SPECIALIZED HARDWARE AND SOFTWARE

The PC-based simulator upgrades all make use of standard, off the shelf hardware and software for functions that, in the past, were performed by proprietary, specialized hardware and software. An example of this would be the use of Microsoft SQL server as a simulator database, replacing a vendor-specific database. Another example is the use of standard third party VME and PCI based systems to handle simulator I/O, replacing vendor-specific hardware solutions. In the past, the specialized requirements of a real time simulator required custom solutions.

The advances in the computing power of microprocessors, as well as the development of standardized high speed I/O busses, have made the use of custom hardware much less necessary. Standard software can now also be used for much of the simulator functionality that previously required custom coding. RNI Technologies has found no instances of a standard hardware or software product used in a simulator upgrade causing a degradation in simulator performance. In fact, the off the shelf solutions are typically faster and of higher quality than the custom solutions.

The major impact that these developments have had on the industry is to increase the competition for upgrades. Small companies, using off the shelf tools, can now compete with large simulator vendors for simulator upgrades. It is no longer necessary to have the capability to manufacture custom hardware. Before 1995, simulator computer platform and instructor station upgrades were uncommon because of the high cost. The PC/Windows NT technology has increased competition and decreased cost, resulting in a large increase in the number of simulator upgrades in the U.S.

PERFORMANCE OF INTEL AND DEC ALPHA COMPUTER PLATFORMS

A major factor driving utilities to upgrade training simulator computer platforms is the need for increased processing
capacity. Simulator software has outgrown the capacity of the original simulator computers, or the utility wishes to implement advanced models on the simulator and requires increased computer power. RNI Technologies has found that Pentium Pro and Pentium II processors compare favorably with the most powerful Silicon Graphics (MIPS) and DEC Alpha processors, and are much more powerful than the Encore RSX. For extremely large simulators, the best choice of processor for Windows NT may still be the DEC Alpha. Benchmark results comparing the 500 MHz DEC Alpha with a 300 MHz Pentium II for typical simulator applications have shown the DEC Alpha to be about 1.7 times faster than the Pentium II. Advanced NSSS and neutronics models from GSE, CAE, RNI Technologies, and Westinghouse have successfully been run on a PC with adequate spare capacity.

**OPERATING HISTORY OF WINDOWS NT SIMULATORS**

As of February, 1997, there are seven simulators in the U.S. training on PC platforms under Windows NT, representing over seven years of operating experience. During 1997, at least six additional simulators will begin training on PC platforms. Table 1 summarizes the status of Windows NT simulator upgrades in the U.S. In Europe, KSU (Sweden) has experience with Windows NT simulator platforms.

**Table 1 - Windows NT Simulator Upgrades in the U.S.**

<table>
<thead>
<tr>
<th>Utility</th>
<th>Simulator</th>
<th>Status (Sept., 1997)</th>
<th>Executive</th>
<th>Upgrade Vendor</th>
</tr>
</thead>
<tbody>
<tr>
<td>Vermont Yankee</td>
<td>Vermont Yankee</td>
<td>Training (Feb., 1995)</td>
<td>OpenSim NT</td>
<td>RNI Technologies</td>
</tr>
<tr>
<td>NSP</td>
<td>Monticello</td>
<td>Training (Jan., 1996)</td>
<td>OpenSim NT</td>
<td>RNI Technologies</td>
</tr>
<tr>
<td>NPPD</td>
<td>Cooper</td>
<td>Training (Apr., 1997)</td>
<td>OpenSim NT</td>
<td>RNI Technologies</td>
</tr>
<tr>
<td>GPUN</td>
<td>Three Mile Island</td>
<td>Training (Jan., 1997)</td>
<td>OpenSim NT</td>
<td>RNI Technologies</td>
</tr>
<tr>
<td>WPS</td>
<td>Kewaunee</td>
<td>In Progress</td>
<td>OpenSim Plus</td>
<td>Yankee Engineering</td>
</tr>
<tr>
<td>PSEG</td>
<td>Salem</td>
<td>In Progress</td>
<td>OpenSim NT</td>
<td>RNI Technologies</td>
</tr>
<tr>
<td>PSEG</td>
<td>Hope Creek</td>
<td>Training (May, 1997)</td>
<td>OpenSim NT</td>
<td>RNI Technologies</td>
</tr>
<tr>
<td>IES Industries</td>
<td>Duane Arnold</td>
<td>In Progress</td>
<td>OpenSim NT</td>
<td>RNI Technologies</td>
</tr>
<tr>
<td>NYPX</td>
<td>Fitzpatrick</td>
<td>Training (May, 1997)</td>
<td>OpenSim Plus</td>
<td>North Coast Software</td>
</tr>
<tr>
<td>Centerior Energy</td>
<td>Davis-Besse</td>
<td>In Progress</td>
<td>OpenSim NT</td>
<td>RNI Technologies</td>
</tr>
<tr>
<td>PECP</td>
<td>Peach Bottom</td>
<td>In Progress</td>
<td>SimExec</td>
<td>GSE</td>
</tr>
<tr>
<td>Duke Power Co.</td>
<td>Catahba</td>
<td>In Progress</td>
<td>OpenSim NT</td>
<td>RNI Technologies</td>
</tr>
<tr>
<td>Duke Power Co.</td>
<td>McGuire</td>
<td>In Progress</td>
<td>OpenSim NT</td>
<td>RNI Technologies</td>
</tr>
<tr>
<td>SCEG</td>
<td>V.C. Sumner</td>
<td>In Progress</td>
<td>OpenSim NT</td>
<td>RNI Technologies</td>
</tr>
<tr>
<td>Union Electric Co.</td>
<td>Callaway</td>
<td>In Progress</td>
<td>OpenSim NT</td>
<td>RNI Technologies</td>
</tr>
</tbody>
</table>

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The simulators that have completed platform upgrades and are training have reported excellent availability since the upgrade. The earliest upgrades, Vermont Yankee and Monticello, experienced some difficulties immediately after completing the upgrade. In the case of Vermont Yankee, most of the problems were due to a parallel upgrade of the NSSS and core models, with some problems due to the OpenSim NT executive software not yet being mature. In the case of Monticello, most of the lost training time during the first few months was due to the simulator engineers and instructors not being familiar with the new system. Additional problems were due to the immaturity of the new instructor station, which was the first full featured Windows NT instructor station, as well as the first true client-server instructor station, to be installed on a training simulator. During these first few installations, many lessons were learned and instructor station and executive software matured greatly.

During the first full year of operation with PC’s, the Monticello training simulator recorded availability of 99.9%. D.C. Cook has reported availability of over 99.9% as well. While the availability at Monticello is at least as good as on the Encore RSX computer that was replaced, they have experienced a greater frequency of simulator computer halts, recording five in the fourth quarter of 1996. The halts are quicker and easier to recover from on the PC platform, however, causing less loss of training time. It is believed that at least some of the halts can be attributed to the relative lack of maturity of the OpenSim NT executive and Instructor Station software (Rev.1 of both are installed on Monticello). More recent simulators that have begun training on PC platforms,

Three Mile Island and Cooper, completed the first several months of training with an availability of over 99.9%. Three Mile Island and Cooper are running with Rev. 2 of OpenSim NT and OpenSim IS, which appear to be much more robust than the earlier versions. To date, there is no evidence that running on a PC platform under Windows NT will result in degraded simulator performance or availability.

The PC hardware, Windows NT operating system, and simulator network hardware and software have not yet caused a loss of training time on any of the seven running training simulators. The hardware and software in the PC environment is perhaps the most mature in the computer industry. The main threat to availability in the PC hardware environment is due to the ability to introduce components from a variety of vendors onto a computer or across a network. This is much less common in a mainframe or workstation setup, where video cards, network cards, hubs, etc., are all manufactured by a single vendor, such as DEC or IBM. In the case of mainframes and workstations, compatibility is not an issue. It has been the experience of RNI Technologies that with PC’s, one cannot be totally confident that a particular combination of components will function reliably. New components should be thoroughly tested in the exact configuration that will be utilized in the simulator system before being introduced onto the simulator. The potential for problems can be minimized by deciding on a video card, network card, hub, whatever, and sticking to it. Compatibility with tremendous amounts of hardware and software from third party vendors is a major advantage of the PC environment, but it is not always as simple as it sounds.
Another lesson learned is that software upgrades of third party software products such as operating systems, compilers, and databases cannot be relied upon to work just because the previous version worked. The problems can be very subtle and not appear for a period of time after installation. RNI Technologies has become very careful about third party software upgrades. There is no need to hurry to be compatible with the latest release of a compiler, database or operating system. The same goes for upgrades and modifications to the instructor station and executive. Run and test any software changes for a while before placing them on the simulator. Never assume that it will work without any problems, no matter how simple it looks. Of course, this is good advice for simulator engineers no matter what computer platform is used.

By far the greatest threat to simulator availability on a PC platform is simulator personnel that are not yet familiar with the new system. It is probably unavoidable that these types of problems will be encountered during the first few months of operation. This can be minimized by maintaining a strong customer involvement in the simulator upgrade and testing program.

ADVANTAGES OF THE PC UPGRADE - REAL OR HYPE?

The advantages that have persuaded simulator owners to upgrade to Windows NT-based platforms include decreased operating and maintenance costs, availability of spare parts, and ease of upgrades. Another advantage is the ability to run the training simulator off-line on standard PC's, allowing testing, troubleshooting and development to take place without occupying the training simulator. This also allows the simulator to be taken into the classroom training environment.

The decreased operating and maintenance cost advantage has been proven, with the removal of Encore simulator computers and dedicated cooling units from service. Upgrade of compilers and operating systems has dropped to a few hundred dollars from several thousand dollars before the upgrade. RNI Technologies has made the transition from Pentium to Pentium Pro to Pentium II and DEC Alpha processors and from Windows NT 3.51 to Windows NT 4.0 with much less effort than similar upgrades on Encore or Silicon Graphics platforms would require.

The third party tools available in the Windows NT environment for coding, debugging and testing code are superior to those available under other operating systems due to the number of developers competing for market share in the PC environment. The volume also makes the cost of these tools much lower than in the Encore or UNIX world.

Advantages due to availability of spare parts has also been proven, with multiple vendors competing to sell PC hardware and software.

Simulator engineers and instructors have taken advantage of the ability to run the simulator stand-alone for testing and development, but so far users have not taken advantage of the stand-alone simulator for classroom training. It will
take more time for training programs to be
developed around this new tool.

A survey of the sites that have been
training on the Windows NT simulators
has shown that users are unanimous in
believing that the upgrades have met or
exceeded expectations.

BIOGRAPHY

Jody Ryan is the President and founder
of RNI Technologies, Inc. Since 1988 he
has been involved with the construction,
testing and upgrade of numerous
simulators. During the last three years, he
has guided the growth of RNI
Technologies into a major provider of
training simulator upgrades and services.

Prior to starting RNI Technologies, Jody
was a licensed SRO and Operations Shift
Manager at Davis-Besse nuclear plant.

Stanley Chan is the Vice President of
Technology for RNI Technologies. In this
capacity, Stan developed the first
Windows NT real time simulator
executive, OpenSim NT, and was the first
person to port a real time training
simulator to Windows NT. Stan was also
the creator of the 'client-server simulator
architecture, and wrote the first Windows
NT simulator instructor station. Prior to
joining RNI Technologies, Stan was a
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GRASS - the Graphic Simulation System

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Abstract

Due to the COCOM restrictions the choice of available computer types was not very rich in Hungary for years. We had to rely on - not very up to date - Hungarian-made clones. The relatively small computer memories and CPU powers determined the style of our simulation tools.

After the removal of the restrictions a few years ago, a very big variety of computers became available. In some industries PC/AT-s are still in use; others keep buying advanced workstations. Our existing simulation tools were unable to meet the demand.

It became obvious that we needed:

- a full-graphic interactive front-end so that the basic modelling could be done by engineers of the customer instead of our simulation experts,
- full-graphic interactive display programs to provide easy testing and verification of the models,
- maximal portability of the products and the development system to very different computers,
- simulation methodology to be optimised for execution speeds rather than minimising memory requirements.

To ensure portability, we have decided to move all our existing simulation tools to UNIX (system V release 3.2) and to develop all the graphic programs with X-Window using the Motif toolkit. The results have been ported to the following computer types: PC/AT486 or Pentium (with SCO UNIX or LINUX), SunSparc 10, IBM RS 6000, DECStation 3000/300x (with OSF1), AlphaServer 2100/4/275 (with VMS and DECWindow).

Our intention was to develop some kind of 'graphic shell' capable to drive very different simulation model solvers. The only restriction is that the simulated process should be defined as a network composed by any number of freely programmable nodes and junctions. In the first phase of development we had to simulate thermohydraulic and electrical networks, plant logic and control and instrumentation circuitry.
1. Modularity and portability

For the shake of modularity we divided the task as follows:

- **Icon editor** and libraries - to create elements with static and dynamic (updateable) graphic parts, with hot-spot areas for interaction, with initialiseing and simulating C or FORTRAN functions, with the declaration of data base requirements;

- **Picture editor** which creates a technological schema connecting the icons mentioned above. The basic output from the picture editor is the generated code for the picture animator, and the directed connection lists and header files for the code generator;

- **Code generator** to generate the simulating (source) code automatically using the icon libraries and the output from the Picture editor. The basic simulating routines are still written by hand and must be well tested before adding to the Icon libraries;

- **Picture animator** which is used to display the values on the technological schema assembled using the Picture editor and to handle human interaction - having access to the data base description files and to the shared memory only.

All the input and output files of all tools are in ASCII and the compilation and linking has to be done as the very final step to be made on the target computer only. Thanks to the easy portability of ASCII files, these tools can be used on different computers even for the same project. For example, the icon libraries can be generated and maintained on a PC/AT, the picture(s) can be created on a Sun computer, the code can be generated on an IBM RS6000 and the final product (the simulating source code and that of the picture animator) can be compiled and executed on a PC/AT again.

The graphical part and the technology-specific part is well separated; exactly the same Icon and Picture editor are used for the logical circuit, thermohydraulic and electrical network simulating code generators. The difference lies in the Icon libraries and in the code generating methodology.

2. Automatic model generation

The task of the model generation is divided into two steps:

- **Icon generation made by simulation experts**
  The icons represent the nodes of the network to be simulated (pumps, valves, heat exchangers, transformers, generators, PID controllers, flip-flops, etc.). The necessary modelling approach is associated to the icons during the icon generation phase;

- **Picture generation to be made by the experts of the customer**
  The actual simulation task is defined by dragging the appropriate icons to the picture, setting their actual parameters and connecting them with other icons.
Fig. 2.1.1. The Software Structure of the GRASS system
2.1. Description of the Functions

The functions are described here according to the numbering of the blocks in Fig 2.1.1.

(1) The Icon Editor
The modelling and simulation capabilities as well as the accuracy and the efficiency of any GRASS application are determined basically by the icons available to the user, therefore the Icon Editor is the most important part of the system. The followings are to be given for each new icon:

- the appearance of the icon on the screen has to be given using the usual graphic primitives (line, polyline, coloured and filled rectangles, polygons, arcs, etc.);

- the input, output, state and constant variables (integer, Boolean, real) for the simulation are to be determined. All input and output variables must have a dedicated connection point on the boundary of the icon.

- a C language code fragment has to be given to initialise the simulation - to pre-set the outputs, the state variables and the constants.

- one or two C language code fragments must be given to advance by one time step during the simulation. For simple modules one routine is usually enough; for modules e.g. with internal storage two program fragments can be given if necessary; the first will be executed at the beginning, the second at the end of each the time step.

- if someone clicks with the mouse to the icon during simulation, the co-ordinates relative to the upper left corner are given together with the code of the push-button to the hot-spot routine which can be associated to any icon, if interactivity is needed.

- for the animation of the picture, the parameters of any graphic primitive can be connected to any variable of the icon by a C language statement. For example, if the beginning co-ordinates of a line are fixed, and the end co-ordinates are calculated using the sin and cos functions from a given parameter of the icon, a turning needle of a panel meter can be constructed. Water levels, different colours for different temperatures etc. can be easily realised using this feature.

The Icon editor is working on icon libraries containing one or more icons. When exiting from the program, three files are updated, shown as (2), (3) and (4) on Fig. 2.1.1. The Icon Editor's window is shown on Fig. 2.1.2. with the of a control valve.

(2) The Icon Description File
A special language has been invented to describe how to draw the face of the icon onto the screen and how to move, turn, increase and decrease the size of different parts of it, how to change the colours according to the actual value of the connected icon variables.
Fig 2.1.2. The Icon Editor

(3) The Icon Header Library File
The input, output, state and constant variables of the icon are collected into C structures and are declared - much like C++ objects - in this header file.

(4) The Source Code Library File
The C language subroutines necessary to initialise the simulation, to advance one time step, to respond to the mouse and to prepare the animation of the icon are collected into this library. All the subroutine and variable names are concatenated from the name given by the user and from the name of the appropriate icon type.

(5) The Picture Editor
The Picture Editor is used to define the real simulation task. At start, we have to open one or more Icon Libraries. The Icon Libraries are constructed usually on a technological basis; separate libraries are made for logic, thermohydraulics, electrical networks, etc. The Icon Libraries contain generic types of the icons. Using the Picture Editor, the real network can be constructed from the icons as from building blocks. Each icon in the picture gets a unique name and a separate data structure to hold the inputs, outputs, state variables etc. of the particular icon. Data base names can be used
freely. If the whole network can not be located on one picture, special icons are used to connect the given picture with another one.

After all the necessary icons have been placed to the picture, the 'wiring' starts. Each connection starts from an output, and finishes connected to one or more inputs. The Picture Editor checks the correctness and completeness of the connections. Not used inputs are connected to defaults defined for the given icon type. Leaving the Picture Editor, several files are created or updated.

The Picture Editor's window is shown on Fig. 2.1.3. with a small part of a thermohydraulic network.

![Diagram](image)

*Fig. 2.1.2. The Picture Editor*

(6) **Checking Data Base Names**
Declaring a particular icon using the generic types we need a unique data base name. It is assumed that the data base has been generated in advance. It is usually not a burden because the lists of necessary elements (valves, pumps) are prepared usually in the very early stage of a project. The particular structure is created always with a valid data base name.
(7) Source File Generated for the Picture Drawing Program (Static Part)
Having all the icon description files (outputs of the Icon Editor), it is easy for the Picture Editor to generate the source code of a program created for drawing all the icons and connections; parts to be animated later are drawn in a way not to confuse the operator (e.g. if the red arrow in the pump symbol means running, green means tripped, yellow means disconnected pump, then this program draws a white arrow which means 'not animated yet'). The program is activated whenever the operator selects another picture to be shown on his/her screen.

(8) Source File Generated for the Picture Drawing Program (Dynamic Part)
After the static part is drawn, the present program maps the shared memory to the actual values of the data structures representing the different icons and the animation (as prescribed by the Icon Library) takes place. This program handles the operator's inputs as well, using the pre-defined hot-spot functions.

(9) Header File with the Icon Declarations
This file is generated for the Code Generator program. It contains all the type declarations (one per icon type) and the actual data structures of the used types (for each icon separately).

(10) Connection List
The list of connections can be simply concatenated for several pictures if they are parts of the same network. The usage of this connection list strongly depends upon the nature of the simulation and upon the methodology used by the model solver.

(11) Source Code for Initialisation
The corresponding calls of the initialising functions are copied here in an actualised way for the particular icons in the picture. This file is used by the Code Generator as well. Moreover, the connections between the icons do not mean any real physical data transfer. Only the outputs are stored, and there are pointers showing that where the other outputs are connected to the inputs of the given icon. These pointers are to be initialised here, too.

(12) Code Generator
Up to this point the header and program files and the data files are created independently from the nature of the simulated phenomena. All the necessary information is provided now for the code generator to prepare the simulation code. The character of the code forming is discussed later on.
Pre-compiler to Resolve Data Base References
All libraries, header files and source codes, even the newly generated simulation code use symbolic (data base) names. The pre-compilers scan the source code and replace the symbolic variable names with the actual ones for mapping to the common shared memory. Up to this point the code is independent from the actual data base structure. If there is any change in the data base, the generation of the system has to be repeated from this point. Obviously, the animation codes must be pre-compiled, too.

Access to Data Base - Get Addresses
Because the type of the real-time data base used is not pre-determined, functions (6) (name exists?) and (14) (resolve address) are required only. The exact form of these calls depends upon the type of the data base system chosen. All the usual data base functions are solved separately and are not included into this system.

Compilers and Linker to Generate the Executable
Our real-time executive requires two routines to perform the simulation: the first to initialise the data structures not stored in the snapshots, the second to advance one (given) time step having all inputs ready in the shared memory and putting all the results there as well. Note that no binary code is used up to this step. This is the first step which is machine dependent - and it is the last step of the code generation.

2.2. The Code Generator Programs

Up to now our system described above is used only for logic circuitry, analogue control, thermohydraulic network and electrical network simulation of thermal (including nuclear) power plants. The actual code generator is very different in each case.

The logic circuitry and the analogue control can be simulated together in a mixed way. Comparators are used to convert analogue signals into logical ones; signal converters, switches, multiplexors etc. to do the opposite.

Both types are simulated in the same way. Starting with elements having pre-defined outputs (as integrators, flip-flops etc.) an optimal solution (calculation) sequence is searched. If there are no loops in the system or every loop contains at least one element with pre-defined output (as it is required for the analogue circuitry), a direct solution sequence can be found and the appropriate list of subroutine calls can be generated. In case of logic networks the loops are iterated until no further signal change occurs. Therefore astable multivibrators are excluded - no oscillations are allowed during evaluation of one time step. (Synchronous flip-flops can be used for such purposes).
The thermohydraulic and electric networks are solved not icon by icon but simultaneously for the whole network, using sparse matrix iteration techniques. The code generated by the system in this case is used only to update the state matrices (calculating the conductivity of each branch) for every consecutive time step.

3. Applications

After several successful small-scale applications the GRASS system is used now for the Reactor Protection System Refurbishment project of the Paks Nuclear Power Plant. During the design phase the new reactor protection algorithms are to be tested and verified thoroughly using the Full-Scope Replica Simulator of the plant.

The following tasks were carried out:

- the existing simulator computer has been replaced to a more powerful one in order to be able to reduce the 1 sec time step of the model execution to 0.2 sec step for the technological models and to 0.1 sec step for the models of the plant logic and C&I simulation;
- the GRASS system has been integrated with the real-time executive and data base utility programs of the simulator;
- all of the existing plant logic and C&I models - including the old reactor protection system's models - are now re-written using the new graphic tools of the GRASS; there were 47 icon types created, and the whole logic is modelled on 767 pictures containing altogether 13697 icons;
- The performance and the fidelity of the simulator has been re-evaluated and verified after a 3-months testing period;
- the first version of the new reactor protection system has been modelled with the GRASS system and tested in stand-alone mode. For this purpose there were 115 new icon types created and the model contains 5267 icons on 232 pictures.

During the autumn the model of the new reactor protection system will be integrated to the existing and already tested GRASS model of the whole plant logic in such a way, that either the old or the new reactor protection system models can be executed according to the wishes of the instructor. The operator training is going on with the old models while in the mean time it will be possible to test the new protection algorithms in normal and accidental situations created on the simulator.

As it is usual, a lot of bottlenecks and drawbacks were detected during this huge work. Several utility programs have been made to enhance the self-documenting features of the GRASS system and to provide more self-checking capabilities. A typical picture from the nearly 1000 is shown on Fig. 3.1.
Fig. 3.1. One of the nearly 1000 GRASS pictures
Using Intranet with Simulation

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Abstract

The rapid development of the Internet/Intranet, has made it possible to apply Client/Server architecture to simulation. It enables the simulator user interface to run in one computer (client) and the simulator software in another (server). They communicate over an Internet/Intranet using predefined protocols. The great benefit is that any computer having a WEB browser installed can be used as a simulator interface.

This paper describes one way to use the WEB concept to retrieve and dynamically update simulator related documents. The documents have to be in a form suitable to the Internet.

There are two servers involved. The first is the WEB server, where the documents are stored. The second is the simulator server, which acts as a provider of dynamic data. The client uses a WEB browser to communicate with the two servers. The protocol for the WEB server is HTTP. The communication with the simulator server is handled by a Java applet, which uses RPC. The simulator server acts as bridge between the client and the simulator. The interface between the simulator server and the simulator is unique for each simulator to be interfaced. A requirement imposed on the simulator is the ability to return the current value of a named variable. A more powerful user interface can be built, if for a given simulator variable, its value can be set. To make a user interface is only a matter of making HTML pages, containing suitable plugins and our Java applet.

In the present study we have made process pictures, using Micrografx Designer. We have written a HTML page, using JavaScript to combine our Java applet and the plugin Quicksilver (Micrografx) making dynamic updates of the process pictures.

It is possible to connect to more than one simulator server, using different instantiations of the Java applet.
1. INTRODUCTION

The abbreviations used in this article are listed below.
CGI: Common Gateway Interface
HTTP: Hypertext Transmission Protocol
HTML: Hypertext Markup Language
RPC: Remote Procedure Call
URL: Uniform Resource Locator

Over the last few years the growth of the Internet has been extensive. Two important things are, the establishment of a communication standard and the development of browsers. Intranet is a local area network using the same standards as the Internet.

A browser, such as Netscape or Explorer, is a program running on the client. It knows how to:
• communicate with other computers
• interpret the HTML commands
• interpret script languages, e.g. JavaScript or Visual basic scripts
• control the execution of user supplied applications.

The user supplied applications need to conform to the requirement of the browser. Plugins (Netscape), ActiveX (Explorer) or Java applets (both) are examples of standards. On the Internet there are a lot of such applications available. We can use a script language to make the different applications interact with one another. Hence we can write a HTML-page, on the server, containing applications and an appropriate script. This page can now be accessed and executed by a browser installed on the client. This feature makes the concept well suited for Client/Server applications.

2. GOAL

To make plant documentation available to clients on an Intranet. It should be possible to update the displayed document with current values from the simulated or real process. The system should be able to handle many clients. These might use different operating systems.
3. USER INTERFACE

The pictures needed are in one of the following categories:
- navigation
- plant circuit drawings
- component data

A browser view of a plant circuit shall display the drawing and a list of the tag names present on the drawing. The functionality required is:
- the current process values shall be shown in the drawing
- clicking on a tag name in the list shall display the component data
- holding the cursor over a name in the tag list shall flash the corresponding component on the drawing.
- to zoom and pan the drawing.

The navigation hierarchy shall be shallow and intuitive.

4. SYSTEM ARCHITECTURE

A Client/Server approach is used, where the clients use a browser to display the plant documentation. The clients are connected to an Intranet. This net also have two servers. The documentation server, configured as a WEB server, has access to the plant documentation. The dynamic data server has access to the process values.

The protocol used to communicate with the documentation server is HTTP. A special protocol, based on TCP/IP is used to exchange information with the dynamic data server. We will use polling to obtain the current process values, to lessen the load on the server.

We will use two plugins, which move the work from the server to the clients. One will handle the display of the drawing and the other the communication with the dynamic data server.

4.1 THE CLIENTS

All pages in the system contain JavaScript and use the HTML format. Some pages have plugins embedded. The "home page" defines a frame set consisting of the general and the main frame. The page loaded in the general frame, contains the plugin used to communicate with the dynamic data server. It will never be replaced. This page also
contains a few buttons, one is linked to the top plant navigation page. The main frame is used to display the requested information.

A plant circuit page defines a frame set consisting of the drawing part and the tag name list part. The drawing part page contains the QuickSilver plugin from Micrografx which displays the drawing. We use its interface functions to display process values and to flash objects. A JavaScript function, which at regular interval requests process values, is present on the page. The request is handled by the communication applet, defined in the general frame. Each tag name in the list part reacts to two events, OnMouseOver and OnMouseClicked. The first will flash the corresponding component on the drawing and the second will send a request for component information. The result of the query overwrites the tag name list. This result page also has a button, which on activation brings back the tag name list.

4.2 THE DYNAMIC DATA SERVER

We assume there is an existing software, either a simulator or a process computer system. To be able to make a server to fit our application, an access function must be present, which given a list of identifiers returns their current values. We further assume that this function is callable from within a C program. Neither of these requirements are severe.

We have a Java applet (client) and an access function (server), each on their own fulfilling the requirement stated above. To make them co-operate, two things are needed:
• communication between the two computers, and
• an interface between the programming languages JAVA and C.

There are alternative solutions. We choose to do the language interface on the client and then use RPC to transfer data between the two computers. The latter is readily implemented. On the server side it supports multithreading, which is needed when several clients access the server simultaneously.

The only problem is on the client side. We have to do the RPC client as a dynamic link library instead of a main program. The main program on the client is the browser. This means that our Java applet has to load the library. A browser will normally reject this request as a security violation. It is however possible to find a way around this problem.

The interface between JAVA and C is normally handled by the JAVA native interface (JNI). Netscape did not support it, instead they have the JAVA runtime interface (JRI), which is very similar. No problems are associated with the language interfacing.
This way of implementing the dynamic data server has little impact on the simulator or the plant computer system. To interface another simulator or plant computer system only a small amount of coding is necessary.

### 4.3 THE DOCUMENTATION SERVER

It is configured as a WEB server. The information present on this server are:
- a few fixed HTML pages,
- plant circuit drawings, made using Micrografx Designer,
- component data, and
- executable programs, used to handle queries and generate the response as HTML pages.

A typical request from a client is handled in three parts:
- interpret the incoming query,
- retrieve the desired information, and
- format the output.

We adopt the CGI standard, for communication between the clients and the server. This means that the URL, sent from the client, contains the name of a program and with the parameters appended. The server puts the parameters in an environmental variable and starts to execute the program. The program will pick up the value of the environmental variable, connect to a database and retrieve the requested data. The final task for the program is to write the result to the standard output in the HTML format. This will be sent back to the requesting client. Plant circuits and component data pages are made this way.

### 5. CONCLUSIONS

This prototype Client/Server system has been successfully implemented. To have a WEB browser executing scripts, which combine different plugins is useful. The information retrieved from a server should be in a form compliant with the WEB standards. In most cases this is not a severe restriction. It is possible to provide plugins as downloads.

An existing simulator or a plant computer system can be made available to the clients. The only requirement is the existence of an access function which, given a list of variable names returns their corresponding values.

The main conclusions is:
WEB technology can now be used to make advanced distributed applications.
Enhanced Productivity of Simulation Engineers

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Abstract

Simulation has always required a hybrid collection of individuals for software development and maintenance, half engineers and half computer scientists. This paper presents a chronology and an indication of some of the technology currently available to simplify simulation software development and maintenance so that engineers can truly be engineers and not computer scientists.

Simulation has always been a black art, requiring individuals with a mixed background to be successful. Prior to the advent of computers, simulation required a mixture of engineers and mechanics to construct and maintain the old Link trainers. When analog computers became available, the mix included engineers, mechanics, and wizards to keep the systems together. Digital computers replaced the old analog gear trains and linkages, but this didn’t necessarily make simulation easier. Then it required engineers and a new kind of wizards - computer scientists.

The technology posed several problems. First, wizards are a scarce commodity. Universities don’t typically offer programs of study in simulation wizardry so you often had to bring in apprentice wizards to study under the masters. Software development became highly personalized, making it difficult for one wizard to fix another wizard’s software. As simulation models became more and more sophisticated, even the wizards couldn’t keep up. By the 1980’s, it became apparent that if simulation was to advance, the process had to be made simpler.

The first leap forward was in executive systems. Executive systems originally provided task scheduling and I/O handling but little else. Software engineers had to keep track of memory management, variable definitions, and revision control. GSE Systems (then known as the Singer Link Company) undertook a project to develop an executive system that would provide an environment to enhance software development. The output of that R&D project was the start of what has become the S3 family of executive systems.

The computer platform of choice in the mid 1980’s was the SEL computer system (later known as Gould and now Encore computers). The operating system for the Gould
computers, MPX, presented one very large limitation to advanced model development - tasks including their variable storage were limited to 128 K words. The common practice of memory management at the time was to use a single large block of shared memory to pass variable information between models. As models became more sophisticated, this task size limitation became a daunting problem. So the first objective of the new executive system was to find a way around this task size limitation. The solution was to break that single large block of shared memory into multiple global areas, each only as large as necessary for the specific task. Rather than requiring the software engineer to juggle all of these memory blocks like a court jester, a central database was used to allocate memory. A set of rules was built into that database to ensure that engineers couldn’t inadvertently (or intentionally) overlap memory areas.

The use of a central database allowed several other features to be added to the executive system to simplify software development. The database tracks all occurrences of a variable in any software module, flagging those modules that require recompiling when a variable definition or constant value is changed. An option is even available to automatically perform that recompilation as a result of a database change.

One tedious aspect of software development is declaration and data statements for variables and constants. This required a lot of housekeeping on the part of the software engineer to ensure consistency between modules. To ease this burden, FSCAN, a software pre-processor was developed. When a software module is submitted to FSCAN for compiling, a header file is constructed with all of the required variable declarations and data statements. It is then joined with the software module that consists of merely the equations and comments, and submits it to the system compiler. Software modules become much easier to read and the software engineer is freed up to do modeling and not memory and data management.

To enhance the environment of software development, a two tiered structure for simulation load control was included in the executive system. An Official Development System (ODS) maintains the official approved training software load while a User Development System (UDS) provides a software development environment. Individual software "users" can extract software modules from the ODS, perform modifications including making database changes, and test those modules either stand alone or along with the other modules from the ODS level. When changes have been tested and approved, user level changes can be merged into the ODS by a system administrator. Software testing can be accomplished as either the real time priority task or in background as a time sharing task to allow software engineers to be productive even when the simulator is being used for purposes other that software development.

This executive structure has become the foundation of over 50 simulators, now available for MPX, VMS, UNIX, and the Windows NT operating systems.

With the software housekeeping issues automated, the next issue was the system models themselves. Engineers still had to be both plant engineers to understand system
operation and software engineers to model the systems. Engineers were breaking up systems into nodes and writing a system of equations at those nodes to model the system performance. The idea was rather than require the engineer to continue the repetitive construction of these equations, why not let the engineer describe the system and allow a model builder to apply the equations and develop a solution.

Due to limitations in computing power, the first model builders were relatively simple. FLOWNET constructed a single phase matrix solution for flows and pressure throughout a fluid system. EDNET performed a voltage and power solution for electrical networks. RADNET performed radioactivity propagation and decay through systems. While these model builders were relatively simple, they did prove the concept of allowing the computer to do the repetitive work and allow the engineer to do the engineering. And the output of the model builders introduced consistency in software coding.

As computers became more powerful and less expensive, it became feasible to employ more sophisticated model builders. TOPMERET was introduced to perform true two phase solutions of fluid systems. Conservation of mass, energy, and momentum in both phases is preserved, and constituents including soluble chemicals, gases, and radioactivity are tracked. ELEGANT was introduced to perform a Norton equivalent circuit solution for electrical systems. Modeling of multiple power sources with proper voltage and frequency response became a simple task. CLASC was developed to perform high fidelity logic and control system modeling. Each of these tools used a standard input file format and the tools provided a uniform software code output that could be integrated with the S3 family executive system. But some wizardry still remained, the input files were not necessarily intuitive for any engineer to read.

The 1990's brought us graphical software for a multitude of applications. GSE Systems applied a graphical front end to our TOPMERET, ELEGANT, and CLASC model builders. Model construction was reduced to an engineer laying out a system on a screen by connecting icons and inputting engineering data to characterize the systems and components. Troubleshooting is performed at the graphical level instead of the software code level. The wizards are replaced with system engineers, that may be wizards in their own rights but certainly not as specialized as in the days of old.

Our latest development in the area of software productivity is Attaché, a graphical monitoring tool for the simulation environment. With Attaché the software engineer can monitor variables in both tabular and graphical format, change constants, and control simulation including run, freeze, snapshot, and reset. Data can even be exported to an Excel spreadsheet for analysis. Gone are the days of the wizards peering at lists of scrolling numbers, looking for an errant glitch.

The wizards may not have been completely replaced. But with the modern modeling environment and tools available, mere mortal engineers can become highly productive simulation engineers without succumbing to the perils of computer science.
Training of Nuclear Power Plant Personnel in the German Simulator Centre

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Abstract

This paper describes the German simulator scene. The simulator training is mainly being performed in a centre. The simulators used are nothing but fullscale simulators and were specified and built in accordance to a common design requirements philosophy. The simulators also support theoretical training by means of data processing devices in the classrooms, being connected to the simulators. As the quality of simulator training depends at least as much upon the instructor’s qualification as upon the simulators, the centre makes every effort to qualify and train their instructors including a final license and the obligation for maintaining competence. All simulator courses are prepared individually according to a common quality standard. The outcome of the course preparation are training materials for the instructors, the trainees and as well for the assessing course observers. The used assessment system is based on an observation of the trainees. Instructors and plant representatives evaluate the trainees’ performance against a detailed set of predefined training goals. It is the Simulator Centre’s continuous effort to optimize all elements of simulator training, the simulators, the instructors and the didactical methods.

1. Introduction

1.1 The Simulator Training Centre

Simulator training for nuclear power plant operators in Germany is conducted in The Simulator Centre in Essen. The companies operating The Centre are KSG and GfS. KSG provides the simulators, GfS performs the training.
The German Simulator Centre is equipped with 14 fullscope simulators in training, of which the latest started training in 1997.

This institution with 160 staff serves 21 nuclear power plant units in Germany, Switzerland (NPP Gösgen-Däniken) and the Netherlands (NPP Borssele) and trains 1,800 persons every year. As a common enterprise the company is owned by 12 utilities, which leads to the necessity to prepare common rules and guidelines for simulator specification, training of instructors, assessment of trainees, training material as well as preparation and methodical running of simulator courses.

![The Simulator Centre](image)

The owners of The Simulator centre, the twelve utilities, defined the following official tasks to be carried out:

- Performing of simulator courses,
- Development of training methods,
- Development of methods for the assessment of trainees,
- Specification, procurement and housing of simulators,
- Maintaining and further development of the simulators and
- Services others than simulator training in the areas such as plant operation, procedures, man-machine interface, theoretical training, etc..

Under this setting of tasks, the Centre runs simulator training as well with own simulators as with such made available by a plant or utility. Of course, the goals are high training standards being achieved in the most price-conscious way.
The operational task of the company is the simulator training, run by 48 instructors. They are supported by service functions, who maintain and program the simulators, who produce the training material or feed the instructors with overall industry's knowledge.

1.2 The German Training Situation

The responsibility for the training of licensed nuclear operators is in the hand of the power plant. This training has to fulfill requirements which are defined in guidelines released by the authorities for

- the initial training of reactor operators
- the licensing of reactor operators and
- their maintaining of competence.

The initial training sums up to an at least 3 years lasting guided training period. It comprises of the training of the basics of nuclear technology, of plant systems and special components training, of on-the-job training on shift and - of course - of at least 7 weeks of simulator initial training.

Retraining is also defined in a way, that - amongst other subjects - the whole contents of the plant's system training has to be repeated consequently every 3 years. Concerning simulator training, 4 simulator course weeks within the 3 years period is the minimum demand. Normal are 2 retraining courses every year.

A PWR-Simulator
The training manager of the plant has a variety of own and national sources to choose from for the design of his training plans. First of all, in all plants exists a training department with a training manager and a lot of technical experts, who are able to present their special knowledge to the trainees. To train the basics of nuclear technology, there are three „nuclear schools“ in Germany with authorities’ accreditation, offering 3 months-lasting theoretical courses. Also, the NPP-vendors and sub-vendors offer a great variety of training on all kinds of technical subjects. Last but not least, there is The Simulator Centre which is responsible for the simulator training of all plants.

Thus, the manager of a power plant’s training department designs his training programs according to his special needs using a broad range of competing institutions.

2. The Simulators

For simulator training in Germany there are used nothing but fullscope simulators. 14 such simulators are available for the training of the personnel of 21 plants at the Simulator Centre’s site.

Simulators of the Centre

<table>
<thead>
<tr>
<th>Name</th>
<th>Simulated Power Plant</th>
<th>Clients’ Nuclear Power Plant</th>
<th>Number of Signals to the Control Room</th>
<th>Start of Training</th>
<th>Manufacturer</th>
</tr>
</thead>
<tbody>
<tr>
<td>D1</td>
<td>Biblis, B</td>
<td>Biblis A/B, Stade, Gösgen-Glattiken</td>
<td>12,900</td>
<td>1977</td>
<td>Singer, USA</td>
</tr>
<tr>
<td>D2</td>
<td>Mülheim-Kärlich</td>
<td>Mülheim-Kärlich</td>
<td>23,400</td>
<td>1986</td>
<td>EA/Singer, USA</td>
</tr>
<tr>
<td>D3</td>
<td>Grafenrheinfeld</td>
<td>Grafenrheinfeld, Grohnde</td>
<td>26,500</td>
<td>1988</td>
<td>Knapp Atlas Elektronik, Germany</td>
</tr>
<tr>
<td>D41</td>
<td>Emshald</td>
<td>Emshald, Neckarwestheim 2, Isar 2</td>
<td>24,500</td>
<td>1996</td>
<td>Siemens/SST, Germany/USA</td>
</tr>
<tr>
<td>D42</td>
<td>Philippsburg 2</td>
<td>Philippsburg 2</td>
<td>26,700</td>
<td>1997</td>
<td>Siemens/SST, Germany/USA</td>
</tr>
<tr>
<td>D43</td>
<td>Brokdorf</td>
<td>Brokdorf</td>
<td>28,600</td>
<td>1996</td>
<td>Siemens/SST, Germany/USA</td>
</tr>
<tr>
<td>D51</td>
<td>Unterweser</td>
<td>Unterweser</td>
<td>16,100</td>
<td>1997</td>
<td>Thomson, France</td>
</tr>
<tr>
<td>D52</td>
<td>Neckarwestheim 1</td>
<td>Neckarwestheim 1</td>
<td>12,500</td>
<td>1997</td>
<td>Thomson, France</td>
</tr>
<tr>
<td>D53</td>
<td>Borssele</td>
<td>Borssele</td>
<td>12,100</td>
<td>1997</td>
<td>Thomson, France</td>
</tr>
<tr>
<td>D56</td>
<td>Obrigheim</td>
<td>Obrigheim</td>
<td>11,000</td>
<td>1997</td>
<td>Thomson, France</td>
</tr>
<tr>
<td>S1</td>
<td>Brunsbüttel</td>
<td>Brunsbüttel, Würgassen</td>
<td>14,800</td>
<td>1978</td>
<td>Singer, USA</td>
</tr>
<tr>
<td>S2</td>
<td>Gundremingen</td>
<td>Gundremingen, B/C</td>
<td>21,800</td>
<td>1983</td>
<td>Siemens, Germany</td>
</tr>
<tr>
<td>S31</td>
<td>Isar 1</td>
<td>Isar 1</td>
<td>15,700</td>
<td>1997</td>
<td>Atlas Elektronik, Germany</td>
</tr>
<tr>
<td>S32</td>
<td>Philippsburg 1</td>
<td>Philippsburg 1</td>
<td>15,700</td>
<td>1997</td>
<td>Atlas Elektronik, Germany</td>
</tr>
</tbody>
</table>

These simulators cover the whole range of operation scenarios which we regard to be „training relevant“. This includes normal and disturbed operation scenarios, malfunctions and incidents as well as beyond design accidents up to a point where core degradation may start (1200 °C of cladding temperature). With these simulators all training goals of initial as well as refresher training can be reached („Full Scope Simulators“).
2.1 The Degree of Simulation

The simulators have been specified according to a special methodology. This methodology enabled us to determine the degree of simulation of each system, subsystem or component. Thus, the simulators are optimized in a way that focus is given to components of the plant which we regard to be relevant and some non-relevant systems or subsystems may be left away.

The methodology for the determination of the scope of simulation consists of three steps:

Step 1: All operation scenarios one can think of are checked whether they are training relevant or not. A set of criteria is used to answer this question: An operation scenario or task is training relevant in case it is
- stressful,
- complicated (difficult to understand),
- sensitive (important) or
- infrequent (and not trivial).

Stressful tasks: During plant operation stressful situations may arise during malfunctions due to many alarms and signals being received in a very short time. Such situations may be difficult to deal with in the case of fast or complex transients and may lead to increased stress.

Complicated tasks: There are operating conditions in a nuclear power plant which are based upon complex knowledge requirements or involve process and I & C systems of complicated design. Such relationships are difficult to impart. A characteristic of complicated tasks is that a fundamental understanding of the theoretical background of the task or of the process is sometimes hard to achieve and it is easy to make mistakes in carrying out the process.

Sensitive tasks: Operating conditions which can result in restriction of operating or availability in cases of maloperation can be described as „sensitive“ or „important“ or „critical“. Such tasks may neither be stressful nor complex but because of their importance they must be trained.

Infrequent tasks: It is normal to forget. Knowledge and skill reduces with time and fades away when there is no repetition or practicing.

Step 2: Each of the four criteria calls for a specific training goal:

If the operator is to react correctly in a stressful situation he must be provided with a procedure for correct behaviour. This procedure must be a strategy -
mainly a sequence of safety related checks and controls - that enables him to filter out the essentials from the flood of information, to analyse it and to initiate adequate corrective measures. It should be the training objective to practice such conditions so thoroughly that the operators will succeed in adopting a correct standard behaviour despite the pressure upon them. The most important exercise here is the post reactor trip behaviour including the check of the safety parameters.

In case of a complicated task it is quite possible to counterbalance such difficulties with operating experience. If the complicated operating condition is repeatedly trained on the simulator, the process will be engraved in the operator's mind. This way, the difficulties due to the complicated nature of the task are compensated by experience through simulator training.

In respect to training objectives for sensitive operating conditions one must be aware that they may not be easy to reach. The training for these is aimed at promoting attention and inducing operators to proceed, in accordance with approved procedures, and not apply their own stylistic variants of operation. The objectives to be attained are attention, concentration, motivation and judgement even for unpopular tasks.

For infrequent operating conditions - in case they are not to be regarded as trivial - it is important not only to train as above but also to refresh the knowledge.

Step 3: For each training goal a specific degree of simulation for this very operation scenario can be defined corresponding to the training goal characteristics.

It is obvious that the training tool for a stressful task must be identical with the real working environment. After all, the goal is a standard behaviour or even a drill. One cannot expect from the operator to apply a lesson learned in a different working environment under the assumed stress without faults. Therefore the simulator needs identity with the real world in the necessary areas.

The same observations are true about complicated tasks. As impressing of operational sequences as a compensation for possible understanding problems is the goal, of course, the actions and the phenotype of the instrumentation must be identical in the real world and in the simulator.

In case of motivational training goals such as support of attentiveness the training tool may be different or should be basically similar.
For infrequent tasks, where the training goal only asks for refreshing of an understanding which has been there before, the training tool may also represent the operational scenario even in a complete different way, i.e. by means of VDUs instead of a real control room.

Summarized one can say that differences between real world and simulator are allowed whenever you want to impart understanding, motivation or attentiveness and you need identity when the training goals ask for the impressing of behavioural sequences.

The Requirements on Simulation

<table>
<thead>
<tr>
<th>Operation scenario is ...</th>
<th>Characteristics</th>
<th>Training Goals</th>
<th>Degree of Simulation</th>
</tr>
</thead>
<tbody>
<tr>
<td>stressful</td>
<td>- psychological Pressure</td>
<td>- Impressing of Standard Behaviour / Strategy</td>
<td>- high</td>
</tr>
<tr>
<td></td>
<td>- Flood of Information in CR</td>
<td></td>
<td></td>
</tr>
<tr>
<td></td>
<td>- Time Pressure</td>
<td></td>
<td></td>
</tr>
<tr>
<td>complicated</td>
<td>- complex, technical and scientific Background</td>
<td>- Compensation of Lack of Understanding by Familiarity and Experience</td>
<td>- high</td>
</tr>
<tr>
<td></td>
<td>- theoretical Understanding not easily achievable</td>
<td>- Impressing of operational Sequences</td>
<td></td>
</tr>
<tr>
<td>sensitive</td>
<td>- Human Error with severe Results</td>
<td>- Motivation to &quot;school-type&quot; correct Behaviour</td>
<td>- basically similar</td>
</tr>
<tr>
<td>in frequent</td>
<td>- never, infrequent or partially experienced in Real-Plant-Operation</td>
<td>- Refreshing, Fixing and Imparting of Understanding</td>
<td>- different</td>
</tr>
<tr>
<td></td>
<td></td>
<td>Presentation possible</td>
<td></td>
</tr>
</tbody>
</table>

2.2 Simulator Features

The simulators of the Centre are designed for the training of all normal and disturbed training scenarios, malfunctions and incidents and even beyond design accidents. This stands for the necessity of two phase flow models in the primary and secondary main circuits, of reverse flow and reflux condenser modes, of fill and vent conditions and of mid-loop operations even with open vessel.

Some operational features of the simulators may be mentioned. All standard components of the simulated plant, as there are mechanical ones (motors, pumps, valves, check valves, heat exchangers, filters, tanks, etc.) and also all 1+1-related standard components (controllers, limiters, transmitters, logic controls, automatic modules, etc.) are modelled including a generic set of malfunctions. Thus, the instructors can manipulate each component in almost all imaginable disturbance modes. The number of possible malfunction scenarios is therefore unlimited.
Another feature of interest may be the new classroom equipment with storage and output devices directly connected to the simulator.

As simulator training is being performed with a daily share of 4 hours of control room training and 3.5 hours of classroom work it is necessary to have the technical outcome of the operation scenarios to the instructors' disposal in the classroom. In the classrooms there are installed up to visual display units and a large screen projection on which stored training data such as transient recordings (up to 1,000) or all process computer outputs can be replayed in real time, slow motion and fast time. Instructors and trainees use the same operational devices to manipulate the outputs in the classroom as are available in the control room. Of course, the classrooms are still also equipped with the traditional devices like overhead-projectors and white-boards to support the technical background discussions by visualization.

The basic Structure of a Simulator

3. The Instructors

By far most instructors have been recruited from universities. Very few are former shift supervisors. Because of this background, the committees of The Simulator Centre installed a guideline for the training, examination and maintaining competence of simulator instructors.
The job requirements for Simulator Instructors

The job-outline of an instructor comprises of three talents: The technical competence, the methodical competence and what we call the social competence, what summarizes the commitment for the company's goals and the dedication to the trainee's individual needs. The first two elements can be transposed quite easily into a systematic training program, whereas the latter one can only be reached by conviction, appreciation and good examples.

The training program for an engineer starting from the graduate university level lasts about three years and is - concerning the technical contents and necessary operating experiences - comparable to a shift supervisor training. In addition, the designated instructor must learn teaching. The work on shift in the real plant requires a minimum time of 26 weeks and the simulator training for the instructors runs up to the triple time span of a plant's reactor operator or shift supervisor.

At the end of the training period there has to be passed a formal examination, which comprises of a written test, observation of a real training scenario and an interrogation covering all aspects of nuclear technology and operation.

To maintain this utility guided license, an instructor must establish proof of at least two weeks of participation in real plant operation and another two weeks of additional technical or didactical training each year.
4. The Simulator Courses

4.1 The Preparation of Simulator Courses

Times are long ago, that it was good enough that an experienced instructor runs simulator training without preparation. A thorough course preparation guarantees that the program is adequate for this very group of trainees, that the training goals do meet the requirements of the plant operation managers, and that a fair and objective assessment is possible.

In the Simulator Centre course preparation for one retraining course series for a plant takes at least 12 weeks of engineering work. The course program is designed in accordance to the three years repetition program.

To fulfill the requirement of repeating all relevant training items within three years, The Simulator Centre listed all training scenarios for a plant and decided there, how often this item should be repeated to be sure of a reliable operation in the real plant situation. This frequency considers the estimated fading away of knowledge and skills for a specific task. It is called the ,,didactic half-life“.

The course program also contains training scenarios about relevant events in the own and other plants, about operational changes of the plant’s design and procedures as well as about individual needs mentioned by the plant management. After having reached a consensus about the program with the plant, the instructor works out a so called course guide, one for each day. In this guide he defines all necessary inputs to the instructor station, the cornerstones of the plant’s response and - as the most important part - the very detailed training objectives, related to each shift member. The latter part of the document is at the same time the set of criteria for the trainees' assessment. In one column the instructor assigns the detailed objective to an assessment criterion.

As the draft of the course guide is a theoretical paper work, the instructor - and in most cases also the training and operational management of the plant - test this course for one week on the simulator: a ,,crash course“. With this method, a release of the training objectives is given by the plant and all people involved in assessment get a common view of what the target is.

Those parts of the course guides, which can be reused as operational modules, are also documented in an ,,exercise guide“. Next time, the instructor must only refer in his course guide to the exercise guide. After some years, hundreds of such exercise guides exist, so that the instructors have a good basis to refer to.
The Exercise Guide

Of course, the preparation of the four hours lasting classroom training is also prepared with subjects definition, with worksheets for the trainees, with work-outs of technical background information and a lot of visualizing technical overhead transparencies. Such items which the instructors think should be prepared beforehand are sent out to the trainees in a „course’s folder”.

Main Steps

<table>
<thead>
<tr>
<th>Time</th>
<th>Instructor's Activity</th>
<th>Content / Process Reactions</th>
<th>Who</th>
<th>Training Goals / Problems / Hints / Target Behaviour</th>
</tr>
</thead>
<tbody>
<tr>
<td>0:00</td>
<td>Terminal</td>
<td>Terminal Instruction Input (check System - Tanks being connected to Ring-Rope 1)</td>
<td>TD</td>
<td>Target Behaviour (VT 1, 3, 5/11) of Tank 3 closed</td>
</tr>
<tr>
<td></td>
<td>In case of System Tank connection to Page 1, the Exercise shall be started as follows:</td>
<td></td>
<td>RD1</td>
<td>TD: Acknowledge of Target Behaviour (VT 1, 3, 5/11) of Tank 3 closed</td>
</tr>
<tr>
<td></td>
<td>Page VT 1, line 16:</td>
<td></td>
<td></td>
<td>TD: TD: Acknowledge of Target Behaviour (VT 1, 3, 5/11) of Tank 3 closed</td>
</tr>
<tr>
<td></td>
<td>Terminal Instruction Input (check System - Tanks being connected to Ring-Rope 1)</td>
<td></td>
<td></td>
<td>TD: TD: Acknowledge of Target Behaviour (VT 1, 3, 5/11) of Tank 3 closed</td>
</tr>
<tr>
<td></td>
<td>Terminal Instruction Input (check System - Tanks being connected to Ring-Rope 1)</td>
<td></td>
<td></td>
<td>TD: TD: Acknowledge of Target Behaviour (VT 1, 3, 5/11) of Tank 3 closed</td>
</tr>
<tr>
<td></td>
<td>Terminal Instruction Input (check System - Tanks being connected to Ring-Rope 1)</td>
<td></td>
<td></td>
<td>TD: TD: Acknowledge of Target Behaviour (VT 1, 3, 5/11) of Tank 3 closed</td>
</tr>
</tbody>
</table>

The Preparation of a Simulator Retraining Course

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To receive usable results concerning the skills and abilities to master each possible operation situation, the methodical approach in simulator training courses aims for real-life-scenarios. Therefore, the 4 hours setting of one training day is a continuous scenario without interruptions, backtracks or new initial conditions. The course subjects are unknown to the trainees, there is always a moment of surprise. No help is given by instructors, the trainees have to heal their own faults. Even more, the instructors create unfavourable operation results whenever they see risk-taking behaviour. Thus, the trainee takes over full responsibility for his actions and therefore agrees upon an objective assessment. Even with a formal assessment system the appropriate way seems to be training in a positive atmosphere, without personal pressure and with senses of achievement.

4.2 The Assessment of Trainees

The assessment of simulator trainees is primarily meant for refresher courses. Here only the control room actions are subjects of assessment.

The result of this is only a contribution to an overall assessment, which the operation management is obliged to fulfil for each licensed operator repeatedly. Therefore, the responsibility, i.e. the final statement, of assessing an operator, is in the hands of the plant - the instructors perform the system and give recommendations.

The assessment system as such is performed in three steps. The first step is the already mentioned course preparation, mainly the definition of the person-related detailed training objectives or expectations. It is of high importance that these objectives are formally released by the plant's operational management. This release is at the same time a quality assurance (QA).

The second step is the observation of the trainees during their work in the simulator control room. Usually part time, i.e. one to three days per week, also a plant's representative takes part in this observation as a second assessor. This observer is prepared either by the QA of the exercise preparation guide (Objectives!) or by taking part in the preparation course. During this observation, each observer documents his findings and tunes them with his colleague. At the end of the course this coordination leads to a unanimous evaluation.

The third step is a talk amongst the observers and each individual trainee. According to the assessment criteria and to the course program, the findings are discussed and - if appropriate - recommendations are considered. No grades, no passing or failing judgements are given. On one format the plant's representative documents wether recommendations were given or not. Of course the trainee gets a hand-out.
Regarding initial training, only very few plants ask for formal assessment because the prerequisites of the trainees are different. Whereas in retraining, the operators continue professionally their normal duties in a different location, the newcomer must establish his skills and knowledge at first. The appropriate way seems to be the training in a positive atmosphere, without pressure and with senses of achievement.

5. Outlook

The German simulator training scene is prepared for the future in so far, that almost all simulators have been built which are necessary for the variety of different plants. Of course, there is going on a permanent optimization of the simulators by implementing plant changes or changes of the operational philosophy. A group of 23 software engineers in the Centre is dedicated to this task, even supported by the suppliers of the simulators.

But installing and maintaining simulators is only part of the training business. Even more important is the development of the necessary training methods and of the prerequisites for a professional implementation of simulator training. After having completed the simulators procurement the main focus will be given to the soft skills of training. As the German Simulator Centre is a unique institution in the country it is necessary to exchange experiences, good practices and new ideas on an international basis. Thus, improvements and training in accordance to state of the art are within reach.
Nuclear Plant Analyzer: An Efficient Tool for Training and Operational Analyses

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B. PEETERS

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ABSTRACT

The advanced computer technology available now at low cost, combined with the maturity of the best-estimate engineering codes are the fundamentals of the Nuclear Plant Analyzer (NPA). At Tractebel Energy Engineering (TEE), the RELAP5 advanced thermal-hydraulics code is used as basis for the NPA that is mainly used for the training of simulator instructors and plant personnel. Using the special graphical features of the NPA, a set of six course modules has been prepared to provide an in-depth physical understanding of the main thermal-hydraulic phenomena that dominate nuclear power plant behavior in normal and accidental plant conditions.
1. Introduction

Among the lessons learned from the few nuclear accidents (Three Mile Island and Chernobyl) and many minor incidents that have occurred, at least two are vital to avoid future mishaps: namely, the importance of a thorough understanding of the plant behavior in abnormal conditions and adequate operator training needed to provide that understanding.

Several developments in recent years provided answers to this preoccupation.

On one hand, in the past two decades, there have been enormous efforts made in thermal-hydraulics research, the findings of which have been synthesized in advanced thermal-hydraulic computer codes (also called best-estimate codes). These codes, which require a lot of expertise from their users and demand very high computer performance, are now increasingly used in the licensing process. Furthermore, they are now considered mature enough to capture the essential physical phenomena in the event of a plant disturbance, and hence to provide a good insight into plant behavior.

On the other hand, during the same period there was a large increase in the orders for full scope simulators, as they are undeniably the best way to train operators, allowing them to cope with a wide range of plant disturbances. Although the fidelity of simulator models has improved continuously, there still exists a gap, at least for thermal-hydraulic simulation, between the simulator models and the best estimate models in advanced thermal-hydraulic codes.

In an attempt to bridge this gap, "nuclear plant analyzers" (NPAs) are being developed that combine the benefits of the high fidelity of advanced thermal-hydraulic codes, and the simulator like man-machine interface required for training. This has become possible now due to the enormous increase in computer performances, at reduced cost.

The basic idea that "a picture can be worth more than a thousand words" was introduced by the code developers who were struggling to interpret the large amount of data generated by the advanced codes. Hence the accent was put on suggestive visual display.

The NPA integrates advanced simulation codes with on-line computer graphics to create an animated display of system behavior.

An NPA, the description of which is given in chapter 2, has been installed at TEE since 1990. Its various applications are described in chapter 3, with a special emphasis on the training program developed at TEE, that is presented in chapter 4.
2. TEE Nuclear Plant Analyzer

2.1 General Description

The TEE NPA [2] is a data driven, interactive graphical display system capable of visualizing a wide range of plant conditions, from normal to accidental plant behavior.

The data are generated by the advanced best estimate thermal-hydraulic simulation code RELAP5/MOD3 [1], developed by EG&G on behalf of the US Nuclear Regulatory Commission (USNRC) recognized and used world-wide for the analysis of nuclear plants thermal-hydraulic behavior.

The vast amount of data generated by the code are processed via the NPA's graphical libraries and displayed on the computer monitor or projected on a screen for classroom exercises.

The NPA is installed in TEE on a HP 9000 Model 735 workstation that may perform interactively typical plant transients at a rate of about three times slower than real time. Pre-defined scenarios may be replayed at a user defined rate that may go from slower than real time in rapid transients, to much faster than real time in quasi steady-state evolution.

2.2 Specific Features

Specific features that distinguish the NPA from other reactor plant simulation visual display systems are:

- The analyzer's visualization module can be coupled to other simulation codes, such as reactor kinetics codes, or severe accident codes for severe accident management, and can be used to display data from a variety of sources, including data files from plant on-line data acquisition systems.

- The NPA is plant-model independent. Plant geometry and functional parameters are specified in the input deck of the simulation code (i.e. RELAP5), and the NPA graphical display is user-defined. Figure 2.2.1 illustrates a synoptic image of Doel 3 adapted for Tihange 2.
Fig. 2.2.1 • DOEL3 adapted for TIIHANGE2

- The NPA is highly portable as it currently runs on several UNIX workstations. It requires ANSI Standard C and FORTRAN77 compilers, and uses several utility programs provided with the UNIX operating system.

3. Scope of Applications

The versatility of this engineering instrument can be illustrated by the following applications.

3.1 Training of High Level Personnel

The product is foremost a training tool for instructors and plant staff personnel needing an in-depth understanding of the complex phenomena that could occur in a plant under different fault and accident scenarios that could progress up to the onset of core damage.

The NPA fully meets the requirements for such training:

- it is fully interactive, and thus enables users to investigate different operator interventions;
it is based on rigorous physical models which are not compromised by the needs of real time calculation, as is the case for full-scale simulator models;

it reflects plant-specific behavior as far as the input model for the base code represents the plant geometrical and functional characteristics. This is an important feature for those utilities operating units with different plant designs;

it represents the results in a pedagogical way by compressing the vast amount of data and displaying it in a concise and pictorial manner, a form more readily assimilated by humans;

For illustration, figure 3.1.1 shows the phenomena associated with loop seal clearing following a small break loss of coolant accident (LOCA) on a specific mask representing the developed length of a primary loop. This visualization enables the trainees to have a clear image of the progression of the intermediate loop draining and the impact of the primary mass distribution, before and after loop seal clearing, on the core uncoverage;

it allows the user to extract and display the time evolution of thousands of variables related to the plant parameters (not readily available on full-scale simulators), in order to identify the precursors that eventually may explain a complex behavior of the plant. The example in figure 3.1.2 gives a list of all the parameters available for one volume; the time history of these parameters may be displayed upon user request.

![Figure 3.1.1 - Loop Seal Clearing Mask](image)

*Fig. 3.1.1 - Loop Seal Clearing Mask*
3.2 Full-scope Simulator versus NPA

Figure 3.2.1 attempts to classify the different types of training tools as a function of scope and depth of simulation. This figure clearly identifies the wide difference in concept between the full-scope simulators (including the Multifunctional Optimized Scope Simulator or MOSS [4]) and the nuclear plant analyzer, and reflects the complementary aspects of both.

Fig. 3.2.1 • Scope versus Depth of Simulation
Furthermore, the objectives of the training tools are different.

Whereas the full-scope simulators and the MOSS are needed for training plant operators for which a symptom- or function-based understanding is required, the NPA addresses more the operation engineers and instructors who need to understand the root causes of the plant response to a wide range of incident and accident conditions. For this purpose, the NPA is capable of displaying a host of local thermal-hydraulic parameters of the plant, which are not readily accessible with other simulation tools such as a full scope simulator.

As the computational power of workstations increases, the likelihood of extending the NPA scope to include the balance of plant also increases. Depending on the complexity of the simulation code, and the depth and scope of simulation of the plant model, real time may be achieved with present technology for most of the accident scenarios of interest. Use of the NPA in this capacity allows state-of-the art simulation programs to be used in applications where computational fidelity has traditionally been sacrificed for speed.

3.3 Validation and Verification of Beyond Design Accident Procedures

Emergency operating procedures for design-basis accidents have been the focus of attention for many years in different Nuclear Steam Supply System (NSSS) owner groups and are tested and validated on plant-specific full-scope simulators. Hence these procedures reflect a state of the art that can be considered acceptable.

The recent emphasis on accident management to deal with preventive and mitigative severe accident measures for beyond-design plant conditions questions the possible use of full-scope simulators, as the plant conditions are usually beyond the limits of validity of the simulator models.

The RELAP5 based NPA is a reliable tool to optimize the beyond-design preventive accident measures such as primary or secondary feed and bleed procedures. Indeed, RELAP5 has been validated extensively on the basis of such accident scenarios simulated in various integral scaled test facilities (i.e. LOBI, ROSA, BETHSY)
Since the RELAP5 code models are validated only for non-degraded core conditions, other codes are now being considered for dealing with degraded core accidents and are being implemented in an NPA environment (e.g. MELSIM based on the USNRC severe accident code MELCOR). At the moment, these codes are being improved and assessed on the basis of various severe accident scenarios, but contain still too many uncertainties to be considered for validation of beyond-design mitigative accident procedures.

3.4 Use of the NPA for Emergency Drills

Potentially, the NPA could be used to provide simulation support in crisis drills that involve emergency exercises at nuclear power plants and the staff of responsible safety authorities. The NPA can be used for preparation and observation of the evolution of the hypothetical accident involving multiple failures, and would provide an accurate picture of the outcome of many of the mitigative actions taken to deal with the accident.

4. Training Applications at TEE

4.1 Principle

Besides a fundamental theoretical training in reactor physics, operation of the technical installations, the content and the use of the procedures, specifications from the final Safety Analysis Report and many others, the operator instructors and NPP staff also receive practical training on full scope simulators where the acquired knowledge is tested.

To have at each moment a full and, above all, correct image of the fundamental processes, the staff must rely on comprehensive and accurate mental models of the physical reality of their installation. The thermal-hydraulic phenomena are an essential part of it.

To acquire this ability, in the past decades several plant models have been derived, such as the STRATEG [3] glass mock-up of a PWR power plant in which mainly the thermal-hydraulic phenomena could be illustrated. However, these training facilities work in a narrow operating window, imposed by the choice of the materials, at very different parameters from those in the power plant. Besides, usually these mock-ups are not really accurate reproductions of a known installation.
By means of the NPA the thermal-hydraulic phenomena of the plant familiar to the staff may be treated in a visual and interactive manner. Teaching on a reliable model of their plant is an important condition to succeed in creating an optimal education environment leading to an optimal training efficiency and to reply on the spot to questions of the trainees.

The NPA gives a new dimension to this training.

The visual interactive learning method bridges the gap between "knowledge and understanding" acquired during theoretical courses and the "practical abilities" gained through full scope simulator training.

The additional gain in fundamental insight in the thermal-hydraulic phenomena is perceived by the training centers as a must for technical advisors and decision makers of the plant.

### 4.2 Course Structure and Feedback

At the request of the two Belgian training centers, SCALDIS for the Doel power plants and Centre de Formation Nucléaire (CFN) for the Tihange site, a training course has been written for the simulator instructors. The topics have expanded since 1990 to cover the modules shown in table 4.2.

<table>
<thead>
<tr>
<th>Module</th>
<th>Description</th>
</tr>
</thead>
<tbody>
<tr>
<td>1.</td>
<td>Heat transfer phenomena</td>
</tr>
<tr>
<td>2.</td>
<td>Two-phase flow phenomena</td>
</tr>
<tr>
<td>3.</td>
<td>Interpretation of the measurements</td>
</tr>
<tr>
<td>4.</td>
<td>Large and small break LOCAs</td>
</tr>
<tr>
<td>5.</td>
<td>Steam generator tube ruptures</td>
</tr>
<tr>
<td>6.</td>
<td>Feedwater line breaks and steam line breaks</td>
</tr>
</tbody>
</table>

*Table 4.2* NPA-supported Courses for Simulator Instructors (1994)

For each module of about 4 hours, the course is structured to:

- repeat the fundamentals needed to understand plant phenomena;
o illustrate, on a specific mask if needed, some typical thermal-hydraulic phenomena, such as critical heat flux, loop seal clearing (figure 3.1.1), etc.;

o illustrate the evolution of an accident scenario, using the synoptic mask of figure 2.2.1, and highlight the relevant thermal-hydraulic phenomena.

The NPA illustrations are generated on-line with the computer, and projected on a large screen by means of a video projector, so that a large group of trainees can follow the computer images on the screen.

For all sessions, the scenarios are prepared beforehand in order to be able to use the non-interactive REPLAY option of the NPA in case the accident evolution is too slow, or the computer "grinding" time is too large. The interactive mode is used to respond to specific questions from the trainees.

It is worthwhile mentioning however, that any session has to be prepared carefully to avoid surprises, or even negative training, since the simulation code requires very careful attention when changing parameters or scenarios.

An evaluation sheet is submitted to the trainees for completion after each session. Besides evaluation of the theoretical content of the course and the pedagogical support, one part deals with evaluating the NPA as a tool for course support.

Figure 4.2 summarizes the responses to five questions by the trainees for the course topics listed in table 4.2.

In the near future, the possibility of implementing non-interactive replays at the training centers for initial and retraining courses for plant operators will be investigated. The use of multimedia techniques is also explored.

One of the recent developments is that, as consequence of the overwhelming positive feedback of the NPA supported training courses, they are also given to the NPP operators.

5. Conclusions

To visualize overall plant behavior as well as detailed complex phenomena for a wide range of accident situations, the technologies needed are now combined in an NPA, i.e.

o best-estimate computer codes,
o interactive graphical display to enhance the man-machine interface,

o high-performance, low-cost, computer workstations.

At present, the NPA can be used for a variety of applications, such as training, validation of plant emergency procedures, and to prepare and follow emergency drills.

The most important and main application at TEE is for training simulator instructors and engineers. On the basis of a training course consisting of six modules, highlighting the thermal-hydraulic aspects, the feedback from the trainees has been very positive.

Use of the NPA as a training aid is expected to expand, by increasing the scope of simulation, by increasing the computer performance to obtain real-time simulation and by implementing multimedia techniques for better man-machine interaction.

At that point, the NPA could become a low cost alternative for full scope simulator training if a full plant replica of the control room is not essential.

References


Multifunctional Optimised Scope Simulators in Central and Eastern Europe

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J.P. Dalleur, J. Houard - BELGATOM, Brussels, Belgium

Abstract

In the field of operator training, multiple functions have to be covered such as basic principles training, training on specific systems, operations training addressing operating procedures in normal, incidental and accidental situations, plant physical phenomena analysis. Training simulators are appropriate tools to meet theses needs. Optimisation of the scope of simulation is required to meet specific training objectives and produce cost-effective solutions that allow for possible future extensions. Training needs and training programs have to be identified with the participation of final users, leading to the development of appropriate training materials: "multifunctional" (also called analytical) optimised scope simulators are a concrete solution to meeting this challenge.

For these simulators, the quality of physical models used is equivalent to that used in the full-scope replica-type simulators. Moreover, all state-of-the-art technical requirements in terms of development of training simulators, must be satisfied: realism of modelling, tolerances, simulated incidents and accidents. Examples of this concept will be illustrated in the paper through the presentation of recent developments of simulators in Central and Eastern European NPPs (VVER-1000, VVER-440, RBMK, BN600, PWR 600).

A brief presentation of the software workshop used to develop these simulators concludes the paper.

1. Introduction

Among the multiple nuclear safety enhancement activities led by the international community in the beginning of the nineties in the countries of the former Soviet bloc, operator training was identified as one of the key issues. There were no state-of-the-art training simulators available for the training of operators. The few ageing simulators were overloaded. The safety culture of the operators and the quality of their training being a key nuclear safety factor, the only way to improve the situation was to rapidly install simulators at individual plants.

Full-scope replica simulators represent an important investment in resources and time. In general, a minimum of three years are necessary to build, test exhaustively and deliver
the simulator to the site. Apart from the significant cost of the simulator hardware and software, additional costs of specific civil engineering works to accommodate the simulator are necessary.

Neither the financial resources nor the necessary time were available taking into account the urgency of the situation. Other, cheaper and faster solutions were therefore searched for.

The solution adopted consisted in a step-by-step development of the simulators. The first step was to develop compact simulators operated from soft panels (reduced cost of hardware, no necessity for special arrangement to accommodate the simulator, no development of the costly control room panels) with a restricted simulation scope, starting with the plant systems which are most important from the safety point of view - the reactor and the Nuclear Steam Supply System (NSSS), the safety systems and important auxiliary primary systems. The Balance Of Plant (BOP) systems can be simplified while satisfying the global mass and energy balances. This approach reduces the cost and the time of the software development, standard off-the-shelf computers are sufficient to run the models. The following development steps can continue with the objective of progressive upgrading of the simulator, while the compact simulator can be already in use on the plant. At the same time, the users provide extremely useful feedback to the simulator developers.

CORYS had extensive experience in the development of compact soft-panel simulators. More than 30 of such simulators were developed for Electricité de France (EdF) to serve as a self-training tool for plant operators. In these simulators the emphasis was put on the KNOW-WHY concept enabling the operator to understand the underlying physical phenomena. The plant systems and transients are represented in a concise and comprehensive manner. BELGATOM developed a screen-operated multifunctional operator for the Tihange NPP in Belgium.

2. VVER-1000 compact simulators

The experience acquired in the field of compact simulators was valorised in the VVER-1000 compact simulator project. Within a period of 12 months a soft-panel operated simulator was developed and delivered to two NPPs in Ukraine (South Ukraine and Rovno) as well as to Novovoronezh in Russia. The simulator contained a sophisticated 5-equation model of the NSSS thermal hydraulics and models of all the important systems of the reactor department. Subsequently the simulator was upgraded to represent the full balance of plant, the model including all important systems of the turbine department, and with a set of didactic images enhancing the understanding of physical phenomena. This simulator was delivered to four sites in Ukraine and to the safety authority (GAN) in St. Petersburg. A version adapted for Unit 5 of the Kozloduy NPP based on the same concept was developed and delivered to the Kozloduy NPP training centre and to the safety authorities of Bulgaria. The VVER-1000 projects were financed jointly by the French government, the European
Commission, EdF and CORYS. An example of a soft panel image of the VVER-1000 simulator is given in figure 1.

![Figure 1: Example of a soft panel image of the VVER-1000 compact simulator](image)

### 3. RBMK-1500 compact simulator

A similar approach was adopted to develop within a restricted period of time a simulator of the main circulation system and the safety-related systems of the IGNALINA NPP RBMK-1500 reactor. The simulator was also developed in two phases responding to similar constraints and requirements as described above. The sophisticated thermal hydraulic model of the main circulation system provides for the simulation of all important accidents including the rupture of an individual technological channel, of a group distribution header, of the main header and of the downcomer pipes. A full soft-panel replica of the plant alarms was also implemented. The project was financed by the EC. The synoptic image of the multiple forced circulation circuit of the RBMK-1500 simulator is given in figure 2.
4. The BN 600 fast breeder nuclear power plant simulator

Another EC-funded project was awarded to the European consortium composed of SIEMENS, CORYS and BELGATOM, with SIEMENS being the consortium leader. The development workshop of CORYS was used as the development basis for the simulator software. SIEMENS is responsible for the development of the BOP models (three turbines, 200 MW each), the hard panels and the I/O system. BELGATOM develops the man-machine interface for the soft panels and the models of the I&C and electric systems. CORYS develops the thermal hydraulic model of the steam generators, the communication software with the hard panels, and supervises the use of its development workshop in the project. An important part of the development effort is carried out by Russian experts from the Institute of Physics and Power Engineering (neutron kinetics, primary and secondary sodium circuits thermal hydraulics, acceptance test procedures) and the company Simulation Systems Ltd. (data package, models of the instrumentation and control systems). The plant experts are also directly involved in the development, mostly in the phases of data collection, preparation of acceptance test
procedures and simulator tuning and acceptance. High fidelity models are used throughout the simulation, including a 3D neutronic code and a 5-equation thermal hydraulic code for the water and steam side of the steam generators.
This simulator is one of its kind. The man-machine interface is a combination of hard panels (for the most important operation functions) and of interactive soft panels (computer screens). A large synoptic panel presenting the main parameters of the simulated plant is also developed. The simulation scope covers all normal operational states of the plant and a number of incidents and accidents. The simulator is currently at the end of its design phase, with the integration starting in October 97. The delivery is scheduled for August 1998.

5. The KRSKO NPP simulator

The KRSKO simulator is a multifunctional simulator designed for training the operating and maintenance personnel of the KRSKO NPP in SLOVENIA (WESTINGHOUSE PWR 600MW 2 loops). Its purpose is the understanding of the physics phenomena involved in the power plant under normal, transient and accidental operating conditions.
The operating procedures allow operation ranging from depressurised cold shutdown with the reactor vessel open up to full power and vice-versa.
The extent of simulation covers all the systems required for operating the plant in normal, incidental and accidental situations. It comprises around 300 plant sensors, 200 pedagogical sensors and 300 actuators.
To reduce the cost and increase the flexibility of the simulator, it is operated exclusively by soft panels with the use of ergonomic, easy to operate interactive images, which represent either soft copies of the real panels or synoptic images of simulated systems.
The simulator is a training tool, where high attention is paid to the pedagogical aspects. For this reason, apart from images directly related to plant operation, a number of "didactic" images aimed to visualise physical phenomena and improve understanding of processes will be developed. Also, the Instructor Station is conceived with the didactical priority in mind. An example of a synoptic image of the main circulation loops is presented in figure 3.
6. EVVEREST project: extended scope multifunctional VVER-440 simulators

6.1. Initial feasibility study

A feasibility study conducted by the EC and concerning the training needs for the VVER-440 units revealed that these could be satisfied in a relatively short period of time and in a cost-effective way by adopting a synergetic approach taking into account the similarities of the VVER-440 units in the different countries. Since the ultimate objective of all the countries and utilities operating the VVER-440s were full-scope replica-type simulators, the requirement was to conceive full scope screen-operated simulators accompanied by a complete technology transfer in order to permit future extensions, upgrades and developments of the simulators and a maximum level of autonomy of the end users and allowing future interfacing with full control room replica hardware. The physical models used were not to differ from those used in full-scope replica-type simulators. The scope of simulation had to cover all the plant systems operated from the main control room of the plant. A pilot model was to be developed for each of the
main types of VVER-440: one for the older V-230 and one for the more recent V-213. The models for the other plants were to be derived from the pilot models. In this way, eight simulators for VVER-440 units at six different sites in five European countries were to be developed:
Russia: Novoronezh Unit 3, Kola Unit 2 and Unit 4
Ukraine: Rovno Unit 2
Bulgaria: Kozloduy Unit 3
Slovakia: Bohunice Unit 2 and Unit 4
Czech Republic: Dukovany Unit 3

6.2. A multipartner, multisite project

In a bid evaluation procedure, the EC entrusted the project to a Consortium of three European companies: CORYS (Consortium leader), BELGATOM and SIEMENS. Subcontractors and end users from all the beneficiary countries were involved in the project. Within the know-how transfer, plant experts participated throughout the duration of the project in model development, integration and acceptance tests of the simulators. CORYS, apart from the overall management of the project and from providing the software development workshop, was responsible for the development of the primary systems and the Instructor Station software; the turbine department systems were designed by BELGATOM, the Man-Machine Interface was developed by SIEMENS. Each of the Consortium partners brought in the project its specific experience and know-how in the domain of simulation and training. The 3D neutron kinetics model is based on the KIKO code developed at the General Institute of Physics (KFKI) in Hungary. The local subcontractors were responsible for the development of the models of the I&C and electrical systems. All the Consortium partners and the Subcontractors used the CORYS simulation software development tools. In this way, this highly complex project brought together VVER-440 experts and simulation technology experts from 9 European countries (Belgium, Bulgaria, Czech Republic, France, Germany, Hungary, Russia, Slovakia, Ukraine). The challenge faced by the developers was big enough to name the project EVVEREST for European VVER Extensible Simulator for Training.

6.3. Technical scope of the project

The simulators satisfy all state-of-the-art requirements for the development of training simulators in the nuclear energy field in terms of simulation scope and limits, realism of modelling, tolerances, simulated incidents and accidents. A three-dimensional neutronic model of the reactor core, a thermal hydraulic model permitting the modelling of nonequilibrium two-phase flows in the primary circuit and all major accidents including the large-break Loss-Of-Coolant Accident, high fidelity models for all the plant fluid systems, electrical systems, instrumentation and control systems. The Instructor Station provides
all the necessary features for efficient, user-friendly and well documented management of the training sessions. The simulator is composed of the simulation computer, three operator stations with three screens each, one shift supervisor station and one instructor station with two screens each. About 100 images are available to the operators to conduct in simple manner the simulated nuclear power plant including more than 20 didactic images providing clearly presented information on thermal balances in normal and accidental situations, on neutron kinetics, mass inventories, etc. Examples of the different types of images are given in figures 4 and 5.

Figure 4: Synoptic image of the Nuclear Steam Supply System of Dukovany NPP

Figure 5: Didactical image of the reactor balances and important physical parameters
The project started in the beginning of 1995 and currently is in the final acceptance phase. The delivery will be accompanied by on-site training of the instructors and of the simulator maintenance staff. The end users will obtain a licence on the use of the full set of the simulator development tools. Together with the extensive know-how acquired during the project, this will enable them to make the simulator “live” and evolve with the reference plant. A recently awarded EC contract relates to the development of simulator training methods for the EVVEREST simulators.

7. The AL94 software workshop for simulator development

The AL94 software workshop is designed for simulator manufacturing, covering all modules of the simulator and all project phases. Its field of application ranges from study simulators through multifunctional training simulators to full scope training simulators. Its design integrates over 5 years of experience in software tool design and more than 20 years of experience in simulator manufacturing by the CORYS company. The main design criteria of the software workshop are:

- Fully graphical easy-to-use environment for all software tools.
- Extensive use of object libraries, optimising development costs and quality of the application.
- Dynamic loading in memory of libraries and data allowing for quick integration of modules without time consuming compilation and linking.
- Integrated environment covering all project phases: data preparation, system design, coding, testing, integration, tuning, validation and documentation.
- Open software design based on standard UNIX (POSIX 1003.4 standard compliant) computer environment and allowing for connection to any other software through shared memory or message queues.
- Management of simultaneous execution on multiprocessor workstations and distributed simulator architecture.

The different modules of the AL94 workshop are connected through the CORTEX run time data base and the ARCHINFO supervisor and communication protocol as shown in figure 6.
Figure 6: The different modules of the AL94 software workshop

The MODELIX model builder is an integrated object-oriented development tool in which the thermal hydraulic models, the electrical models and the I&C models are developed within a unique user-friendly graphical interface and can be connected with each other only by using the same object names within each model. The system models are designed and tested without any compilation or code generation. In an integrated simulator, the same development environment is used to display any set of model variables, modify them if necessary and to tune the models. An example of a part of the turbine steam system developed with MODELIX is shown in figure 7.
7.1. Object libraries

The software workshop is based on the extensive use of object libraries. This has the following advantages:

- Centralised development and validation of objects by experts,
- Maximum reuse of models over several projects,
- High configurability and modularity as objects can be combined and extended,
- Clear association of code, variables, parameters and documentation in one object.

The development of objects is done with the object editors of MODELIX or DRO, the source code is written in C or C++ in an integrated development environment offering "make" facilities, version management and code analysis. The data structure of each object is conserved in the shared memory of the integrated simulator and can therefore easily be identified, analysed or tuned during all phases of project development. The objects are stored in libraries which group objects of the same type and simplify the management of the objects.

The development process of an object is sketched in figure 8.
7.2. Modular development with AL94

AL94 has important features for modular development and testing. A convenient environment for modular development allows for:

- Independence of different developers in big projects,
- Use of small sized workstation during development and test phase,
- Pre-integration tests by assembling stepwise modules without changing the environment,
- Tuning of steady states and snapshot generation module by module. Snapshots can be used after integration,
- Stand alone tests of entire plant systems as integration of I&C, hydraulic and electric models can be processed in the MODELIX tool.

This is achieved by the following advanced concepts:

- Use of the same object oriented data structure and memory allocation in the stand-alone environment and the integrated simulator. Snapshots are directly associated to the object data structure and can be easily updated after changes,
- Use of integrated configuration and execution software in the MODELIX tool. This is achieved by a consequent use of dynamic memory allocation and dynamic library link to object code. This concept allows for going back and forth between configuration and testing by simple button press,
- Use of powerful **analyzing software** (configurable curves and trends with file or paper documentation) and **interactive parameter control** during simulation,
- Use of **pre-integration features of MODELIX** allowing for assembling testing and tuning of I&C, hydraulic and electric models within the same tool,
- Use of special tuning software as **steady state finder** which is integrated in the workshop
- Use of a **scenario feature** which provides a well defined and reproducible test environment.

### 7.3. Integration methodology in AL94

Integration is clearly the key step in simulator development which brings with it some intrinsic constraints on the methodology to use:
- A constant quality level of incoming modules has to be verified,
- Modules ready for integration have to be delivered with a well prepared set of initial data,
- Progressive integration should be possible,
- Swapping back and forth of modules between integrated simulator environment and stand alone environment should be easy and efficient, as corrections and modification have to be expected,
- The same analyzing and tuning software should be used in the integration environment and the stand-alone environment.

The AL94 software workshop has been designed and continuously extended especially to cope with the above mentioned constraints:
- The advanced model configurators with integrated test environment as presented in the previous chapter enable to define and verify the model quality to any degree of completeness. Standard test sets with generated documentation can be fixed and run through,
- The use of standard object oriented data structure based on CORTEX guarantees full compatibility of initial data generated during development phase and integration phase,
- The ArchInfo supervisor can disable and enable modules dynamically without restarting the simulator,
- Integration of a new module needs no linking or compilation of object code. A restart of the simulator with a new configuration file is sufficient,
- The configurator graphical interface can be used for result display and parameter control.
8. Conclusion

Multifunctional simulators have proven to be an extremely useful tool in operators training, complementary to the use of full-scale control room replica simulators. Their major advantages are:

- short acquaintance time, user friendly,
- efficient KNOW-WHY training of different categories of plant personnel (study of physical phenomena),
- easy implementation of additional didactic images and pedagogic support features,
- can be used directly at or near the NPP site,
- can be used in classroom environment, gives trainees the possibility to analyse the status of any system variables and their combinations, including variables not displayed in the control room,
- low investment and operation costs in comparison with full-scale replica simulators.

Due to the modularity of the development software and to its user-friendly nature, incremental development of simulators is possible, meeting a number of existing constraints:

- limited budgets and often complex funding schemes,
- adaptability to the required delivery times with the possibility of incremental development,
- transfer of know-how and mastership of the development tools by the experts of the plant training centres,
- easy upgrading of the simulators which can follow the evolution (often frequent) of the real plants.

Acknowledgement

This paper could not have been written without the precious skills and tremendous effort of many people with a wide range of expertise in the complex field of simulator development. Not having the possibility to name them all, the authors wish to express their deep gratitude and admiration to all of them.
Leningrad NPP Full Scope Simulator - New Generation Tool for Training and Analysis

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Abstract

Recent developments of Russian Research Center “Kurchatov Institute” are aimed at the soonest and minimum expenditure’s creation of modern high-performance means for personnel’s training and regular re-training and also at the development of means of modeling, safety analysis and operators’ support with severe accidents to be included. In this view, the most elaborated one is the Training Support Center (TSC) created at the Leningrad NPP, Sosnovy Bor, Russia. To be applied in this center, GSE Systems of USA and RRC “Kurchatov Institute” have jointly developed the Total Training System that incorporates full-scope simulator; analytical full-scope simulator, expert system, interactive system, psycho-physiological system; and training support programs. Mathematical models creating and special software development were the responsibility of RRC “Kurchatov Institute”, the hardware and general purpose software were the responsibility of GSE Systems.

1. RBMK-type Reactor and Power Unit as the Objects for Simulation

A principal design of the first channel-type graphite moderated reactors had been implemented in the USSR in late 40-ies - early 50-ies, mostly as reactors for Pu isotopes production for military use.

The specialised design of commercial channel-type boiling reactor with graphite as moderator RBMK-1000 (1,000 MW(e)) had been firstly implemented at the Leningrad NPP from 1974 y. (1st unit), till 1981 y. (4th unit).

There are now fifteen (15) RBMK-type units under operation at Russia, Ukraine, and Lithuania.

The RBMK-type unit’s main features are such that provide a lot of obstacles and difficulties for its modelling and full scope simulation, e.g.:

- coolant’s boiling and saturated steam generation at each of the 1,600 pressure tubes (fuel channels), axially positioned inside the graphite blocks of the huge core (7 m height and 12 m diameter);
- non-stable neutron flux and power distribution and possible positive reactivity effects, that required distributed flux monitoring and reactivity control with using a number of sophisticated automated systems of in-core distributed detectors, of local power distribution control, and of the operator manual control by each of 220 control rods, that could be autonomously manually and automatically driven to be positioned on different height of the core;

- permanent "on-line" refuelling at a full power (2-3 fuel channels per day);

- an absence of primary and secondary circuits as well as of a steam generator in its "common sense" traditional meaning. Namely, a saturated steam and water mixture from each core channel is to be combined and accumulated at four drum separators, and, after that separation, the saturated steam passes onto two turbines with electrical generators 500 MW(e) each. But the remaining water from the drum separators as well as the turbine condensed water are passed back to the group header collectors and further to the core channels by means of feedwater pumps and main circulation pumps;

- the core neutron flux and power distribution instability and a need in a special distributed automatics control system to cope with it,

- a need in a complex and sophisticated Plant Process Computer (PPC) system (SKALA) to control the unit. Importance of SKALA is such that under its malfunctioning the unit should be immediately passed to decreased power levels and later passed to shut down,

- a need in very sophisticated neutronics and thermal hydraulics codes to support as the design and operation, as well as the simulation.

The a.m. RBMK features provide with complicated and detailed 3-D tools for the core modelling (neutronics and two phase thermal hydraulics), sophisticated tools and codes for the Balance-of-Plant, and, last but not least, solving the question for the modelling (simulation vs. stimulation) of the Plant Process Computer (PPC) "SKALA".

There are additional difficulties for the full scope simulation of the operators' activity at the working places as well as.

The RBMK-type unit is operated by the team located at the Main Control Room (MCR) and consisted of the senior supervising operators as Plant Shift Supervisor (PSS) and/or Deputy Plant Shift Supervisor (DPSS), and of three executive operators as the Reactor Operator ("RO" to monitor and control core neutronics and thermal hydraulics, neutron flux and power distribution, reactivity, state of the Reactor Safety and Protection systems, reactor's heat and steam production, etc.), Unit Operator ("UO" to control balance-of-plant), and Turbine Operator ("TO" to control turbines, electrical generators, electrical distribution and electrical devices, etc.)
In addition, there are Local Post Operators positioned at Local Control Rooms and Local Control Posts in different areas of the plant to control and supervise, e.g., Fuel Reloading Machine, Water Chemistry, Plant Electricity Dispatch Center, Reactor Main Circulation Circuit and Pumps, Turbine and Generators Central Hall, Radiation Monitoring and Environment Protection, etc.). These Local Post Operators are communicated with the MCR Operators by a number of lines.

The Customer Project specification required with the complex joint training of the whole Operation Team, as of the Main Control Room, as well as of the Local Posts.

So, the operation of the RBMK-type units is rather complicated and would require a set of specialised measures to be simulated with.

In addition, the PPC "SKALA" which is the main instrument to help for operators is rather obsolete as for hardware as well as for its software and Man Machine Interface (MMI) design of early 70-ies, and is under permanent up-grade and modernisation during last years. New tools, devices and systems for operators' support are under development, testing, and implementation. Since, it was demanded to provide with the "open" possibilities for SKALA algorithms simulation to be up-graded and changed.

This task was solved with the development of a special tool for SKALA codes emulation.

And, at last, there was a requirement of modelling RBMK severe accidents at the Customer Project Specification for the Simulator. Putting in mind the Chernobyl Accident and very sensitive attention to the RBMK up-grade and its future fate as from the Regulatory Bodies, as well as from the International Communities, mass media, publics, etc., this requirement is seemed as fully reasonable and timely one.

2. Main Features of the LNPP Full Scope Simulator

Factor of risk caused by the personnel faults primarily in non-regular and emergency situations may be rather great. In world practice the major way of its diminishing is the application of simulators. It betokens the RRC “Kurchatov Institute” experience in creating the full-scope simulators, analysers and large modelling complexes for Nuclear Power Plants (NPPs) and other nuclear facilities.

Recent developments of RRC “Kurchatov Institute” are aimed at the soonest and minimum expenditure's creation of modern high-performance means for personnel’s training and regular re-training and also of means of modeling, safety analysis and operators' aid with severe accidents to be included. In this view, the most elaborated one is the Training Support Center (TSC) created at the Leningrad NPP, Sosnovy Bor, Russia. To be applied in this center, GSE Systems of USA and RRC “Kurchatov Institute” have jointly developed the Total Training System that incorporates:
- full-scope simulator;
- analytical full-scope simulator;
- expert system;
- interactive system;
- psycho-physiological system;
- training support programs.

Mathematical models creating and special software development were the responsibility of RRC "Kurchatov Institute", the hardware and general purpose software were the responsibility of GSE Systems.

LNPP full-scope simulator enables maintaining and training with operational personnel the correct actions' skill in real time under the assigned emergency situations' conditions and with involving in the training process of the enlarged shift's staff. It is possible to apply not only the full-scope Main Control Room (MCR) operational prototype (MCR-O), but non-operational MCR (MCR-N) panels too, as well as the local posts "soft control panels" computer models.

LNPP full-scope simulator utilizes the high-capacity I/O system to ensure data exchange and display on the MCR panels and on the color graphic displays in 27,000 parameters.

There are three (3) instructor stations included in the full-scope simulator and located at the MCR for providing effective training process.

The simulator provides the new scope for the user:

1. The possibility of holding training for the entire shift of 50 persons by employing six remote work stations as the operations' reproduction from the local control posts.

2. The portion of work stations' displays has a form of main and local CR "soft panels".

3. The hereabove Items 1 & 2 enable the reproduction of real mistakes and/or faults of the personnel that arise during training.

4. The component malfunctions allow to perform practically infinite number of variable and emergency situations with severe accidents to be included.

5. Development and evaluation of the prototypes of the personnel support systems (Operator Aids) is supported by the simulators' technology which enables the monitoring of the training process by the application of special formats.

6. There is a possibility of local posts' operation as jointly with the simulator as well as separately in off-line mode.
It's the first time ever that the full scope and analytical simulators ensure the modeling of the severe beyond the design-basis accidents, making the appropriate technology processes' analysis and trying out provisions. Therefore, these new-generation simulators are apt for training the personnel to act in severe beyond the design-basis accidents, and also for the development and evaluation of appropriate scenarios to provide emergency training as well as for development and evaluation of the Operator Support Aids, e.g. SPDS-type and on-line expert systems.

With a view to modeling, the RBMK-type reactors are among the world's most complicated ones. The 3-D mathmodel of RBMK utilized in the LNPP full-scope simulator has as its base the most detailed and verified codes developed in RRC "Kurchatov Institute":

(a) the advanced STEPAN_SIM 3-D neutron kinetics "real time" model to calculate fuel channels with various degree of enrichment and burnup, all kinds of absorbers and control rods. The delayed neutrons incorporate 24 groups for U-235, U-238, Pu-239 and Pu-241. Neutron cross sections are the functions of fuel temperatures, coolant density, fuel burnup, etc. The simulation option of the code is based with the STEPAN off-line engineering code.

STEFAN model and code are verified by the wide set of static and dynamic reactor experiments. Thus, STEPAN full-scope neutron-physical module, integrated with KOBRA full-scope thermal-hydraulic module to incorporate the distributed mathematical description of all the RBMK 1661 fuel channels of multiple forced circulation circuit enabled first time ever to reproduce complete and non-contradictory pattern of the Chernobyl accident development initial phase as based on data actually registered at Chernobyl NPP unit #4 during the Accident.

STEFAN code is based on realistic 3-D representation with no simplifications as synthesis, symmetry, etc.

(b) the advanced KOBRA_SIM thermal hydraulics "real time" model, which is thermally unbalanced and mechanically non-homogeneous (steam and water). The KOBRA_SIM is based with the KOBRA engineering code as the best estimated model. This model utilises so called "first principle approach" with a full-scale map of modes, reliable constitutive correlations and properties of water and steam through the entire range of steady-state and emergency modes;

KOBRA thermal-hydraulic code may be applied for the calculation of non-stationary processes in the random system of interconnected steam generating channels with heat exchange. This code allows to make calculations for a wide range of the modes parameters (including the steam superheated up to 1,000 ATM pressure) and is used for modelling various nuclear power units dynamics.
Numeric methods utilised in KOBRA code are based on the direct solution of mass, pulse and energy conservation equation. KOBRA code allows to calculate non-stationary processes in an unbalanced approximation.

First in world practice it is capable of describing not only interconnected neutronics and thermo-hydraulics processes, but also the integrated with them thermal-mechanics distortion & destruction processes in the nuclear reactor core and the impact & disturbances introduced by automatic control, protection devices, the operators of MCR and other control posts.

(c) the advanced STALACTITE thermal mechanics "real-time" model, which is used for calculations and stress analysis of the fuel rods, pressure tubes (channels), for core elements melting, distortion & destruction processes, etc.

Thanks to the integration of the (a), (b), and (c) integrated systems, first in world practice it becomes capable of describing not only inter-connected neutronics and thermo-hydraulics processes, but also the integrated with them thermal-mechanics' distortion & destruction and melting processes in the nuclear reactor core and the impact & disturbances introduced by automatic control and protection devices, by the operators of MCR and other control posts.

The LNPP full-scope and analytical simulators are also used to evaluate the safety and efficiency of engineering approaches to maintenance, modernization or equipment replacement, that is, upgrading process systems or control & protection systems, operators support computer systems (expert "advisers"), and also maintenance and emergency provisions and any other MCR improvements prior to their accomplishment at the real power units.

Introduction of similar simulators will make it possible to significantly improve NPP safety, as well as the qualities of its specialists training and to diminish the possibility of accidents rise.

The LNPP simulator creating work meets the advanced criteria for full-scope simulators' development, creation, integration and testing.

American National Standard “Nuclear Power Plant Simulator for Use in Operator Training ANSI/ANS-3.5.” is used as the basic standard.

The section pertinent to modeling of the severe accidents employs the major provisions specified in IAEA documents and the RF NRC ("Gosatomnadzor") recommendations:


The Recommendations by the Federal Supervisory Body on Radiation and Nuclear Safety (Gosatomnadzor) for the OPB-88, Requirements for Beyond the Design-Basis Accident Management. 1990.
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Leningrad NPP Full Scope and Analytical Simulators As Tools for MMI Improvement and Operator Support Systems Development and Testing

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Abstract

Training Support Center (TSC) created at the Leningrad NPP (LNPP), Sosnovy Bor, Russia, incorporates full-scope and analytical simulators working in parallel with the prototypes of the expert and interactive systems to provide a new scope of R&D MMI improvement work as for the developer as well as for the user. Possibilities of development, adjusting and testing of any new or up-graded Operators' Support System before its installation at the reference unit's Control Room are described in the Paper. These Simulators ensure the modeling of a wide range of accidents and transients and provide with special software and ETHERNET data process communications with the Operators' Support systems' prototypes. The development and adjustment of two state-of-the-art Operators' Support Systems of interest with using of Simulators are described in the Paper as an example. These systems have been developed jointly by RRC KI and LNPP team.

1. Reasons To Use LNPP Simulators for MMI Improvement and Operators' Support Systems Development and Testing

A practical need of the MMI up-grade and inclusion of new Operators' Support Systems into the operation of the existing NPPs is evident under the new demands, requirements and regulations. Of utmost importance is this goal for the NPPs with the RBMK-type reactors, such as the Leningrad NPP ones.

A principal design of the first channel-type graphite moderated reactors had been implemented in the USSR in late 40-ies - early 50-ies, mostly as reactors for Pu isotopes production for military use.
The specialised design of commercial channel-type boiling reactor RBMK-1000 (1,000 MW(e)) had been firstly implemented at the Leningrad NPP from 1974 y. (1st unit), till 1981 y. (4th unit).

There are now thirteen (13) RBMK-1000 units under operation at Russian NPPs (Leningrad - 4 units, Kursk - 4 units, and Smolensk - 3 units) and Ukrainian NPP (Chernobyl - 2 units), and two (2) RBMK-1500 reactors (1,500 MW(e)) at the Ignalina NPP (Lithuania).

The RBMK-type units' main features are such as:
- coolant's boiling and saturated steam generation at each of the 1,600 pressure tubes (channels) of the huge core (7 m height and 12 m diameter);
- neutron flux and power distribution, and reactivity control with using a sophisticated system of in-core distributed detectors and by each of 220 control rods, manually and automatically driven;
- permanent "on-line" refuelling at a full power (2-3 fuel channels per day);
- an absence of primary and secondary circuits as well as of a steam generator in its "common sense" traditional meaning. Namely, a saturated steam and water mixture from each core channel is to be combined and accumulated at four drum separators, and after that separation the saturated steam passes onto two turbines with electrical generators 500 MW(e) each. But the remaining water from the drum separators as well as the turbine condensed water are passed back to the group header collectors and further to the core channels by means of feedwater pumps and main circulation pumps.

The RBMK design has its benefits:
- easy obtain of large unit power,
- lack of a huge pressure vessel,
- factory production of core and unit components,
- on-line refuelling;

But it has losses too:
- lack of the real containment,
- the core neutron fluxes and power distribution non-stability and a need in a special distributed automatics control system to cope with it,
- a need in a complex and sophisticated Plant Process Computer (PPC) system (SKALA) to control the unit. Important that under malfunctioning of SKALA the unit should be passed to decreased power levels and later shut downed,
- a need in very sophisticated 3-D neutronics and thermal hydraulics codes to support as the design and operation, as well as the simulation.
The RBMK-type unit is operated by the team located at the Main Control Room (MCR) and consisted of the senior supervising operators as Plant Shift Supervisor (PSS) and/or Deputy Plant Shift Supervisor (DPSS) and of three executive operators as the Reactor Operator ("RO" to monitor and control core neutronics and thermal hydraulics, neutron flux and power distribution, reactivity, state of the Reactor Safety and Protection systems, reactor’s heat and steam production, etc.), Unit Operator ("UO" to control balance-of-plant), and Turbine Operator ("TO" to control turbines, electrical generators, electrical distribution and electrical devices, etc.)

In addition, there are Local Post Operators positioned at Local Control Rooms and Local Control Posts in different areas of the plant to control and supervise, e.g., Fuel Reloading Machine, Water Chemistry, Plant Electricity Dispatch Center, Reactor Main Circulation Circuit and Pumps, Turbine and Generators Central Hall, Radiation Monitoring and Environment Protection, etc.). These Local Post Operators are communicated with the MCR Operators by several lines.

So, the operation of the RBMK-type units is rather complicated and would require a set of measures, mostly to improve MMI and enhance safety features under normal, abnormal and accident modes in a view of the new RF and International Regulations and Recommendations.

Unfortunately, the PPC "SKALA" which is the main tool to help for operators, is rather obsolete as for hardware as well as for its software and Man Machine Interface (MMI) design of early 70-ies, and is under permanent up-grade and modernisation during last years. New tools, devises and systems for operators’ support are under development, testing, and implementation.

E.g., a lack of the information display system at the MCR working places of the senior supervisor operators (PSS, and DPSS) is a serious obstacle to control the main technological process and to provide timely measures for the supervisor operators during the actions’ insurance and support of the executive operators under emergencies and accidents, especially severe ones.

More of that, significant efforts to up-grade RBMK Safety Design and implement a set of additional Safety Systems are provided by the Russian Project and Design Institutes and International Organisations and Communities. These measures and new designs should be tested, validated and verified with using any different possible vehicles - experiments, theory calculations, modelling and simulation.

Training Support Center (TSC) created at the Leningrad NPP, Sosnovy Bor, Russia, incorporates full-scope and analytical simulators in parallel with the prototypes of the expert and interactive systems which provide a new scope of R&D work as for the developer as well as for the user. Possibilities of development, adjusting and testing of any new or up-graded Operators’ Support System (e.g., SPDS-type) before its installation at the reference unit Control Room are the most important ones.
So far as these Simulators ensure the modelling of a wide range of accidents and transients and provide with special software and ETHERNET data process communications with any prototype of the Operators' Support systems, therefore, these new-generation simulators are apt not only for training the personnel to act in emergencies and even at severe accidents but also for the development and evaluation of the appropriate scenarios to provide emergency training as well as for development and proof of the Operator Aids.

The LNPP full-scope and analytical simulators are also used to evaluate the safety and efficiency of engineering approaches to maintenance, modernisation or equipment replacement, that is, upgrading process systems or control & protection systems, operators' support computer systems ("expert advisers"), and also to maintenance and emergency provisions and any other Control Room improvements prior to their accomplishment at real power units.

The development and adjustment of two state-of-the-art Operators' Support Systems of interest with using of LNPP Simulators are described in the Paper as an example. These systems have been developed jointly by RRC KI and LNPP team.

2. ODES (Operational Diagnostics Expert System) Prototype

ODES is provided for operational "on-line" automated diagnostics of malfunctions at the NPP basic process equipment and for intellectual support of operators. The ODES is a further state-of-the-art development of the approach had been successfully implemented by Dr. N.N. Lebedev in two Expert System codes:

DIAG code which was firstly commissioned at the Smolensk NPP on late 80-ies - early 90-ies. The code is running under MS DOS at the separate PC to be communicated with the Plant Process Computer "SKALA". It is used for the NPP basic process equipment malfunctions' operational diagnostics and for the intellectual support of the operators and, if necessary, of the premises shift personnel. Moreover, DIAG code may be used for output of on-line data for the management and R&D specialists, as well as for training and knowledge check of the operational and maintenance personnel.

SPRINT code which is under successful pilot operation at the Ignalina NPP, Unit #1 (Lithuania) from early 90-ies. The code is running under SM operating system at the PPC "TITAN" and with its environment. At present, within its scope are 27 main process systems fully covering the power unit (with 60 subsystems in total). Over 10,000 malfunctions are subject to diagnostics, and upward 30,000 rules are installed in the data base / knowledge base.

By now, the unique experience with using of the Expert System (ES) codes DIAG and SPRINT at the operating NPPs has been accumulated as practice of work with experts.
(technology engineers and operators) as well as of power unit's behaviour in transients and non-regular situations data process recording and filing. On the other side, the tools themselves, the operating systems environment and coding, as well as the technology of the ES design and the "knowledge base machine" execution became rather obsolete and didn't satisfy with the operators' and practice demands and needs.

The situation and position of the old Expert Systems is more aggravated with the plans on PPC (SKALA and TITAN) up-grade onto the new state-of-the-art computer platforms and technologies.

So, the new ODES tool and code's development technology have been developed and implemented at the LNPP Simulator. To meet the state-of-the-art high demands to the ES tool it was decided to use as a base platform of the ODES the new AIS-95™ ("Automated Interactive System") approach. The ELUD (Easy to Learn, Use & Develop) ideology has formed the basis for AIS-95™ and Expert System development. (See the Paper "New Technologies of Modelling and Simulators Creating" by V.V. Shaila at al. at this 2nd Meeting).

The ODES system proved to be particularly effective at:

- avalanche data flows;
- diagnostics of control equipment and automatics;
- operators' errors;
- onset of violations.

Present ODES is aimed at on-line diagnostics of the NPP basic process equipment malfunctions. The foundation of ODES is formed by expert models of diagnostics, which comprise database of measuring process parameters, knowledge base of parameters behaviour, assorted texts of diagnostics messages and recommendations for the operator. To make reference for the elements of the basic process equipment, ODES is provided with Network On-line Data Base (NODB) consisting of measured power unit parameters' current values.

ODES system creating is aimed at increase in safety reliability and power unit operation effectiveness by:

- automated (with pre-set time interval) early-stage detecting of malfunctions and breaks in all the basic process systems;
- output of diagnostics results for the operational personnel in concise and suitable form;
- output of recommendations (advise) for the operator on confining malfunctions and trouble-shooting (as per the current process regulations, technical manual and other specifications);
- diagnostics results and adjoining process data archiving on computer storage media.

Fulfilment of works is tied with algorithms and diagnostics codes creating; analytical and experimental study of static and dynamics mode characteristics of basic nuclear power unit equipment, optimisation of diagnosis process and its confining advice messages results visualisation with optimisation of the man-machine interaction (MMI).

Scientific and experimental background for this study are diagnostics algorithms synthesis techniques and programs, calculations of static and dynamics, techniques and programs of experiments, operating rules, programs of overhaul repair (OR) and maintenance repair (MR).

ODES should enable the following functions:

1) iteratively, with pre-set time interval to sample NODB for obtaining data on diagnosed process systems state;

2) iteratively, at each sampling through the acquired information to detect abnormal states in each diagnosed process systems;

3) iteratively, at each sampling, for each detected abnormal situation by expert diagnostics model to determine the reasons of abnormal states (situations) emergence;

4) at regular intervals to archive data on the hard disk on abnormal states, reasons of their origin, as well as recommendations for their confining;

5) to output to the operator data on abnormal situations, reasons of their origin and advise on their confining with indication of the kind of process equipment they pertain to;

6) to output by the operator's (or process engineer's) demand an archive information on abnormal situations, reasons of their origin and an advice on their confining with the indication of the process equipment they specifically pertain to;

7) through the dialogue with expert process engineer to carry out data and knowledge base sampling and processing: input, editing, testing and comprehensive processing of power unit process subsystems expert models.

ODES prototype is being developed and implemented to provide communication with the LNPP Full Scope Simulator (reference Unit # 3). However, ODES could be regarded as stand-alone "off line" system and should provide the possibility of its operation both with the simulator and with the real PPC of the operating power unit as well as. For this reason a modular approach was standardised. E.g., the communication protocols have been designed to be based with standard TCP/IP, and so on.

ODES is running under the standard Windows NT 4.0 and up at any PC as Pentium, Pentium Pro, and up. Since some PPC and Simulator Host computers are running under
UNIX, the communication system has appropriate protocol features for on-line data exchange under different Operating Systems.

ODES simulator-mode operation envisages the possibility of conducting scientific study on elaborating diagnostics techniques and algorithms and on use of the expert system as a mean for "on-line" operators' support and MMI improvement.

As a representative part of the plant diagnostics the "Turbogenerator # 2 vacuum sub-system" was selected for careful investigation and testing under different simulated normal and emergency operation. This sub-system incorporates the interconnected by the main technology process parts of equipment of the following plant systems:

- FW - feedwater,
- MS - main steam,
- OG - off-gas.

The experiments at the FSS with the given ODES sub-system have demonstrated its profound and stable "on-line" communication and operation with the successful recognition of 95% of appropriate malfunctions.

By now the rest systems and sub-systems of the unit are under development as diagnostics part of ODES for its further testing with the FSS. It is assumed, that after testing of ODES as a part of the simulator and after all the necessary modifications the system should be installed at the real power unit # 3 of LNPP to be on-line communicated with the PPC SKALA..

3. GOSS (Generalised Operator Support System) Prototype

The first priority goal of GOSS is to decrease of mistakes' probability and, especially, of personnel' "wrong decisions" during NPP operation.

The GOSS priorities are to be based upon a concept "System of Dynamics Barriers (SDB) to prevent a propagation of Accident Transients at the NPP" by the same authors. GOSS is oriented, at the first, toward a support of operators under "stress conditions" and is based on the hierarchical tree structure of display pictures with the root display picture "Safety Observation Screen" (SOS) where all the information needed for control of the main technology process and enough for a safety monitoring, is to be displayed. The SDB methodology allowed to solve such principal questions as:

- types' classification of power unit's conditions (states) in accordance with the IAEA recommendations, from the 1st (normal) through 2nd and 3rd (incidents), 4th (accident "scram"), and even down to 5th (project design accident), 6th and 7th (severe accident);
- determination of the operators' main function and of supporting system's one as the function of the power unit condition's type identification during transient, incident or accident;

- choosing an optimal package of "videogrammes" (display video screens) for reliable supervising control of all types of power unit's conditions;

- choosing the content and the information presentation optimal ways, based on the condition's type, time of transient and character of tasks of the plant shift supervisor operator;

- determination of spheres (areas) for emergency procedures of different types (I, A, H1-4, U1-5 and others), worked out by International communities.

The special technology to display an information (with a computerized processing for its generalization to be included) is developed to provide a possibility for a "global view" to the object is being controlled, as well as for definition of dangerous states / modes and the most important control functions.

Foundations for GOSS development are:

- requirement of the RF, NRC (GOSATOMNADZOR) "to include for NPP control such a System for an operational displaying of a generalised information on the safety state and mode of NSSS and of NPP as a whole";

- wide-spread concept on development and perfection of the "Instrumentation Tools" for Operators as the first priority mean to increase a "human factors" reliability;

- lack of any "Information Display System" (IDS) at the Main Control Room working places of so called Supervising Operators of the RBMK (Plant Shift Supervisor - PSS, and Deputy Plant Shift Supervisor - DPSS).

The aims of the GOSS development work are:

- creation of an advanced Operators' Tool to increase quality of a unit control based on a new approach to content, means and formats of an information displaying;

- development of the Support for Supervisors to organise an additional safety barrier of operation supervising control.

GOSS is oriented, at the first, toward a support of operators under "stress conditions" during fast transients or in a case of possible severe consequences, when all the information needed for control of the main technology process and enough for a safety control, is to be displayed.

GOSS concept is a generalised concept and picture imagined hierarchy model of the unit (from NSSS up to the electricity distribution and output) oriented for the working places of Supervisors. In this case Supervisors could obtain a possibility to monitor all the states (modes) of the unit and control actions of Operators (executors). It gives an opportunity "from the first glance" to monitor fast transients of the unit as a whole,
provided with in-time revelation of faults and automatics protection failures as well as violation of limits, pre-sets and conditions of a normal operation. Simplified and unified interface allows to obtain manually by request) or automatically all the needed information.

Other words, GOSS should be the universal and "all-modes" one, with an unified package of display frames (diagrams) to support all the operators' activities for any mode (regime) of the unit. Main diagram, as the root display picture "Safety Observation Screen" (SOS) is being displayed permanently at the main CRT of the Supervisor, reflects all deviations in the unit's operation which are important for the process' reliability and safety. Under accident conditions the special video diagrams to control three "critical safety functions" should be automatically activated. In a case of a threat or an explicit violation of the safety functions the special code for a search of workable channels to execute each of the safety functions should be initiated.

Features of the RBMK-type unit as the object to be controlled were put into consideration in the development and testing process of GOSS for the 3rd unit of the Leningrad NPP:

- a lot of enunciators, alarms and indication lamps for different parameters being randomly displaced at the "traditional" Main Control Room and, correspondingly, a huge volume of a badly organised information, especially under transients;

- a lack of the Project designed IDS for Supervisor's working place;

- a necessity of a prompt co-ordination, supervising, and "insurance" for three so called "executive operators" by the Supervisor.

The main user's features of the system are:

- providing of a generalised information on unit's operation;

- simple and unified instrument for a needed information obtaining;

- comfortable perception of an information;

- initialisation of operators actions to parry deflexions and malfunctions.

As an example we refer to the "Safety Observation Screen" displaying for the GOSS of the 3rd unit of Leningrad NPP which is under evaluation and adjusting at the Full Scope Simulator. The following functional nodes are presented: pump sets, reinforcement units, main regulators, safety system executing devices. This one SOS picture contains a lot of information:

1. Condition of limits and sets of safe operation.

2. Readiness to work and the safety systems' work ("in", "on", "off").

3. Monitoring of "margin" for triggering by analogue parameters of protection for reactor and turbines.
4. Parameters' condition of main technological process.
5. Condition's control of main automatic regulators of power unit.
7. Control of organized leakage and feed of coolant.
8. Control of aqua-chemical state of power unit.
9. Control of the integrity of physical barriers of radioactivity.
10. Control of feed supply for unit own steam and electricity self-needs.
11. Blocking and automatics failure / malfunction control.
12. Detailed information of the second level. It is used during shift's accepting-leaving procedures. It is switched automatically off in case of the emergency states arising.
13. Extinguishing of fire protection system condition.
14. Information about an outside influence (flaring up, flooding, explosion, natural calamity).
15. Text information ("hot line" string of a priority information), namely:
   - Duplication (by text) of emergency automatics failure;
   - Providing of urgent communication with remote posts in case of failure of the main channel.

GOSS prototype is being developed and implemented to provide communication and parameters exchange with LNPP Full Scope Simulator (FSS). However, GOSS, as well as ODES (see above) could be regarded as stand-alone off-line system and should provide the possibility of its operation both with the simulator and with the real PPC of the operating power unit as well as. For this reason a modular approach was standardised as well as the communication protocols, list of parameters, and so on.

GOSS prototype is running now under the standard UNIX at the INDY workstation or at the host FSS computer CHALLENGE (Silicon Graphics Int.) with X-Windows terminal displaying and under FSS environment with ETHERNET networking. A successful use of the base GOSS approach for the generalised description of the main technology process and of the unit's systems and sub-systems had allowed already to use GOSS hierarchy of the display formats as the upper three-level process presentation for the Instructor Station and for the Analytical Simulator operators' interface development.

Since GOSS should be installed later at the real operating power units to be communicated and adjusted with the unit's PPC, and since some PPC computers are running under UNIX, and others - under Windows NT, so the GOSS communication system has appropriate protocol features for on-line data exchange under different Operating Systems.
In addition, a new option of GOSS software to be run under Windows NT is under
development now. This option shall be based with the same base platform as of the
ODES, namely the new AIS-95™ ("Automated Interactive System") approach. The
ELUD (Easy to Learn, Use & Develop) ideology has formed the basis for AIS-95™ and
for ES and new GOSS development. A joint platform's use would provide a
compatibility of both systems with their possible merge and more wide options for their
use at the plants.

GOSS simulator-mode operation have allowed already to provide research and scientific
study on elaborating the picture imagined hierarchy model of the RBMK unit and
algorithms of the unit's modes (states) recognition as well as on use of the system as a
mean for "on-line" operators' support and MMI improvement.

The experiments at the FSS with the given GOSS prototype have demonstrated its
profound and stable communication and operation "on-line" work with the successful
recognition of the unit's states and operators' support up to the 5th level ("design"
accidents). The work with the severe accidents' operation modes recognition and
operators' support would demand its further development and continuation as for the
GOSS algorithms as well as (last but not least) for the FSS itself detailed severe
accidents' modelling, adjusting, verification and validation.

By now the systems and sub-systems of the RBMK-1000 (LNPP unit # 3) are fully
covered with the GOSS three-level picture images. The work for a permanent operators' 
support for each unit's state (mode) is under development as the most important part of
GOSS for its further testing with the FSS. It is assumed, that after successful testing and
validation of GOSS as a part of the simulator and after all the necessary modifications
the system should be installed at the real power unit # 3 of LNPP to be communicated
with its PPC.

Introduction of similar simulators' use as testing devices for the operators' support
systems will make it possible to significantly improve NPP safety, as well as the
qualities of its specialists training and operators' support, and to diminish the possibility
of accidents' rise.

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Abstract

The simulator-based Halden Man-Machine Laboratory (HAMMLAB) has, since its establishment in 1983, been the main vehicle for the human-machine systems research at the OECD Halden Reactor Project. The human factors programme relies upon HAMMLAB for performing experimental studies, but the laboratory is also utilised when evaluating computerised operator support systems, and for experimentation with advanced control room prototypes. The increased focus on experimentation as part of the research programme at the Halden Project, has led to a discussion whether today’s laboratory will meet the demands of the future. A pre-project concluded with the need for a new laboratory, with extended simulation capabilities. Based upon these considerations, the HAMMLAB 2000 project was initiated with the goal of making HAMMLAB a global centre of excellence for the study of human-technology interaction in the management and control of industrial processes.

This paper will focus on human factors studies to be performed in the new laboratory, and which requirements this will bring upon the laboratory infrastructure and simulation capabilities. The aim of the human factors research at the Halden Project is to provide knowledge which can be used by member organisations to enhance safety and efficiency in the operation of nuclear power plants by utilising research about the capabilities and limitations of the human operator in a control room environment.

1. Introduction

The Halden Man-Machine Laboratory, HAMMLAB, was established in 1983 in order to serve as the main environment for performing realistic experiments within the Man-Machine Systems Research of the OECD Halden Reactor Project (HRP). Since its establishment, HAMMLAB has been the experimental focal point of the research within Human Factors, as well as the main test bed for computerised operator support systems being developed both at the Halden Project and at members organisations.

The NORS full-scope simulator has since the establishment of HAMMLAB been the laboratory’s simulator basis. NORS is based on the Loviisa nuclear power plant in Finland (1).

HAMMLAB has undergone major upgrades and improvements since 1983, the last one being performed in 1996 with the introduction of a new unified human-machine interface and a new control room set-up (2,3). The upgrades have partly been made to support the Halden Project’s research programmes, partly due to specific requirements set forth in bilateral funded experiments and studies. The Halden Project has
experienced an increased demand for a facility able to support advanced human factor's related experiments, both through members organisations in the joint research programmes, as well as through requests for doing specific studies for certain organisations on a bilateral basis. With this in mind, the Halden Project wanted to investigate whether today's HAMMLAB would be able to meet tomorrow's needs for an experimental facility, and the HAMMLAB 2000 project was initiated.

2. HAMMLAB as of 1997

The studies being performed in HAMMLAB are of different size and complexity, ranging from large scale human factors experiments to small scale studies and tests. Due to this, the requirements to the laboratory vary a lot, and a flexible infrastructure is a necessity. The current control room is equipped with two operator stations and one supervisor station, as indicated in Fig. 1. All stations are situated on desks having wheels in order to ease shuffling around and varying the degree of compactness of the control room. In this way it is extremely easy to restructure the control room for one or more operators. Another key issue is that all information is available on all screens, thus allowing for tests using single operators and few screens.

![Fig. 1 The HAMMLAB Control Room](image)

The carrying through of a large human factors experiment requires careful preparation prior to the actual execution of the experiment, and a large period for data analysis after the experimental execution. The data collection phase, i.e. the actual execution of the experiment, requires the availability of advanced data recording equipment. Audio and video recorders, eye movement trackers, computerised data logs of various kinds, are heavily used in addition to questionnaires and on-line expert commenting. HAMMLAB of today provides the experimenters with advanced data recording equipment and everything is configured and operated from a specially designed experimenters' gallery.
Fig. 2 shows a picture of the experimenters' gallery where the human factors experts carefully follow what is going on in the HAMMLAB control room.

3. The HAMMLAB 2000 Research Agenda

The HAMMLAB 2000 project was initiated by means of a pre-project early 1996. The goal of the pre-project was to try to predict tomorrow's research needs within the area of Man-Machine Systems research, and point out requirements to an experimental facility based upon future needs. Such requirements should then eventually lead to a comparison with today's HAMMLAB to see whether today's laboratory met tomorrow's demands.

3.1. The Basis for the Research Agenda

When discussing advances in human-machine systems research, the technological development is fast and to some extent unpredictable. On the other hand, the main characteristics of the human being does not change in the time scale of interest. The human capacity for perceiving, thinking, and acting develops so slowly that it for all practical purposes may be considered constant. This makes the prediction of future research needs for controlling complex processes a little bit easier, because it must concentrate on development and adaptation of new technology, starting out from the characteristics of the human being.

In order to make any predictions, it is necessary to make some basic assumptions, and the following assumptions are considered pertinent (4):

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• People (humans) will remain an essential part of the control of complex dynamic processes. In other words, there will not be full automation. Sub-systems may become fully automated, but they will always be embedded in a larger system.

• The scope of control will increase in terms of the system boundaries and the time horizon. The system boundaries will grow and technology will enable processes to be coupled over significant distances. The control will be of the coupled process rather than of the local process which may be highly automated and control need no longer be confined to a central control room. The time horizon must be extended to ensure a sufficient economy. It is necessary to plan not only for the current situation, but also for the future, e.g. after a change of process parameters. Down-time must be reduced and availability increased, involving aspects of preventive and state-based maintenance. Safety will, however, still be in focus both on the short-term and the long-term scale.

• There will be a greater need for predictions as part of control, hence a need for modelling and simulation. This follows partly from the extension of the system boundaries and time horizon, but also from the increased speed and complexity of processes. Prediction is an essential function both for the process control operator and for managers, and for operation as well as e.g. maintenance (outages), procurement, decommissioning, design, safety assessment, etc.

• The design of MMI has on the whole been driven by technological innovation rather than by sound human factors principles or user needs. Despite a growing awareness of the importance of human factors, it is probably safe to assume that the development of MMI will to a large extent continue to be driven by technology. Since people will be the same with their recognised strengths and weaknesses, this will perpetuate the known problems and probably even create some new ones.

3.2. Possible Research Themes for the Future

Based on the assumptions made above, and the current research being made within human factors and human-machine interaction, a prediction of possible themes for future research can be made. Some of the themes are continuation of current research, some deduced from today’s research. As a basis for the future research, two specific items are considered important to take into account:

- Team Co-Operation, Distributed Work

It is a truism that a person never works alone, but despite that, most experiments have looked at the single operator. It is time to relinquish that restriction. Furthermore, the
technology encourages decentralisation and distributed work. Research topics are collaboration, communication, information sharing, co-ordination between people and technology, regardless of physical and temporal locations.

- More Realistic Work Settings

There have been two different schools in human factors research: the well-controlled laboratory experiment and the field study. Both have provided useful results, but they have sometimes been in apparent conflict in terms of the concepts and theories they have used. The revision of the information processing basis means that there is less emphasis on strict experimental control, and it may therefore be an appropriate task for an experimental facility to demonstrate that the two can be combined. This will lead to experiments - or “simulated field studies” - with a higher degree of realism (longer scenarios, more realistic tasks, more naturalistic and complex work settings, work in teams), to supplement the more conventional hypothesis testing experiments.

Based on the issues and trends discussed in this chapter, one can argue for a human-machine systems research agenda which includes at least the following items.

- Developing A Joint Situation Understanding

Developing a joint situation understanding among people who are in the same room is not straightforward. This has led to considerable research for e.g. training and large overview displays. When the scope of the system grows larger, particularly when the boundaries of the system extend to other locations, the problem of establishing, supporting, and maintaining a common situation understanding becomes very much larger.

- Information Integration

There will be a significant need to be able to integrate information from various sources, regardless of the original formats or protocols. As the system boundaries expand in space and time, it is necessary to include data within the new boundaries. For the operator, there will be a requirement or a need to look at a particular combination of data in order to assess the situation. The system technology should be able to provide this within a reasonable time limit and with a minimal effort from the operators' side.

- High-Level Automation

The effects of high-level automation in the system design should be studied in an experimental environment. The software of the high level automation will use input from the process, the normal automatic systems, the protection systems, the
computerised operator support systems and the operators. The automation can either execute the resulting high-level control commands directly on the process and the normal automatic circuits, or present them as suggestions/recommendations to the operator, who will then take the final decision whether to execute the control commands or not.

- **Accident/Emergency Management**

There is an increased interest for research in the field of Accident or Emergency Management. This work aims at utilising the capabilities of computerised tools to support various users such as the control room staff, people in the technical support centre, and safety bodies providing advice to national authorities. The use of experimentation in a simulator environment should be very interesting in order to improve the management of emergencies (5).

- **Integration Of Maintenance Activities And Process Surveillance**

There is a strong trend towards increased integration of maintenance tasks and process operation tasks. This obviously has consequences for several aspects such as:

- integration of maintenance staff and operation staff
- the ability to prepare, train and even perform several maintenance tasks during power operation
- integration of computerised tools for maintenance with systems used during normal operation.

4. **The HAMMLAB 2000 Project**

The predictions of future areas of research, some of them mentioned in the previous chapter, will impose requirements to the experimental environment. In addition, direct transferability of results, i.e. to make studies with the use of a process similar to an organisation's real process, whether it is a nuclear power plant of BWR, PWR or VVER or the petroleum industry's oil & gas production process, is given more emphasis from member organisations of the Halden Project. Some studies are, however, generic in nature and the results are likewise applicable across industries, therefore synergy is also an interesting aspect being focused upon by several organisations.

Some of the main requirements to a future HAMMLAB is listed below:
• The process simulators must be easy to configure and efficient tools must exist to implement different types of nuclear power plants and conventional processes.

• One particular important area of human-machine interaction research is the sharing of tasks between operators and automation. As a consequence, the digital control system must be easy to reconfigure and provide means for varying the degree of automation.

• Various operator support systems such as alarm analysis, diagnosis, and prognosis, will be further developed utilising new methods and techniques. This will require new high-level implementation tools which will facilitate reuse of software modules and knowledge when considering different processes. The goal is to reduce the time required for changing a system from one process to another. In principle, only the plant-specific knowledge should need modification.

• The hardware infrastructure must be designed to take care of new software development, integration of new software, perform experiments, and carry out the experiment analysis.

• Finally, the central control room and adjacent facilities such as several local control rooms, experimenter facilities, observers’ gallery, project rooms, data and documents storage facilities, demo/info centre, etc., must be established to ensure sufficient space and equipment for carrying out the proposed research activity in an efficient way.

Comparison of the above requirements to the functionality of today’s HAMMLAB, indicated that today’s HAMMLAB is a good basis, but would not meet the requirements to an experimental environment of the future.

This conclusion led to the start-up of the HAMMLAB 2000 project with the goal of establishing a new HAMMLAB meeting tomorrow’s demands. The HAMMLAB 2000 project has the goal of establishing a flexible infrastructure regarding the physical laboratories and the hardware and software, as well as making sure tomorrow’s HAMMLAB has a broad pool of simulators. The latter will be described somewhat in more detail below.

4.1. The HAMMLAB 2000 Simulators

As part of the HAMMLAB 2000 pre-project, requirements to process simulators to be included in the new HAMMLAB was identified. The requirements are briefly
summarised below. In addition, this section gives an overview of the planned simulators for HAMMLAB 2000, see Fig. 3.

**Current Situation**

- NORS "westernised" VVER

**Wanted Situation**

- CP0 Fessenheim PWR
- NORS "westernised" VVER
- Forsmark-3 BWR
- Oseberg Petro

*Fig. 3 The HAMMLAB 2000 Simulators*

### 4.1.1. Simulator Requirements

The process models are the heart of the simulator facility, and its quality determines to a large extent the applicability of the results obtained in human factors and systems research. In this context quality refers to various properties of the plant model. The main items of importance for determining the quality and applicability of a process simulator are listed below.

- **Scope**

  The simulator shall be full-scope, i.e. all systems in the process relevant for control room operators shall be simulated.

- **Simulation Range**

  The simulator must be able to simulate the process realistically for a large set of normal, disturbance and accidental states.
- **Process Realism**

The simulator shall simulate in detail a given process. This is very important with respect to: documentation, verification and validation of process behaviour, and training/use of experienced operators in human-machine interaction experiments.

- **Malfunctions**

A wide range of malfunctions shall be available for use in experiments and tests. This includes malfunctions for all types of components and systems. In addition, it must be easy to make new malfunctions. It is also required that the simulator and its environment shall be able to handle instrumentation errors and inaccuracies.

- **Process Control System**

The overall process surveillance and control system shall be designed to or somewhat beyond the state-of-the-art for modern industrial plants. The implementation solution shall resemble modern plants as much as possible, with potential for significant improvements.

- **Automation Systems**

It must be possible to include all types of automation systems in use in industry today and planned for the next 5-10 years. This includes modern distributed control through utilisation of PLCs. In the simulator, the process and automatics shall be clearly separated, making it easy to implement independent changes in process and automatics.

- **Real Time Performance**

The simulator shall be able to run in real time over the whole simulation range with realistic process behaviour.

- **Integration Of Operation And Maintenance**

The plant simulator shall be prepared for being coupled to something like a maintenance planning simulator. The latter will be used to plan the maintenance work in detail, in order to minimise shutdown time and costs. Furthermore, a version of the plant simulator may in the future be extended to become a life cycle simulator for preventive maintenance. This tracking simulator will run in parallel with the real process and generate automatically on-line updated maintenance parameters.
4.1.2. The NORS Simulator

The NORS simulator is a “westernised” VVER simulator of the Loviisa nuclear power plant in Finland. NORS was manufactured in 1983 by Nokia Electronics, and has since then been the nucleus of HAMMLAB. Several modifications and additions have taken place of the NORS simulator models, and NORS is now considered to be a very good simulator for the purpose of being “the process” when performing experimental studies. Although HAMMLAB 2000 will consist of several new simulators, NORS will play a role also in the coming years as part of the HAMMLAB 2000 pool of simulators.

4.1.3. The CP0 Fessenheim Simulator

The member organisations of the Halden project have clearly expressed the wish for a western type of PWR simulator as part of the HAMMLAB 2000 pool of simulators. After discussions with different simulator vendors, the Halden Project is currently negotiating with Thomson Training & Simulation and Electricité de France about the delivery of a full-scope simulator of the Fessenheim-1 plant in France. Fessenheim-1 is a Westinghouse-like 900 MW 3-loop plant built by Framatome. Depending upon a successful finalising of the contract, the simulator is scheduled for delivery to Halden in the summer of 1998.

4.1.4. The Forsmark-3 BWR Simulator

In order to prepare for maximum transferability of results from experimental studies, it has also been decided to include a simulator of a BWR plant in HAMMLAB 2000. BWR utilities in Sweden and Finland have expressly stated their interest in having a BWR simulator as part of HAMMLAB 2000, since they see a clear benefit to be able to run specific studies in HAMMLAB to feed results into their large control room modernisation programs.

It has been agreed between the Swedish and Finnish utilities and the Halden Project to make a BWR simulator based upon the Forsmark 3 power plant, a 1160 MW BWR located close to Stockholm in Sweden. The simulator will be manufactured by VTT Energy in Finland in co-operation with the Halden Project, and is scheduled for completion late 1998.
4.1.5. The Oseberg Simulator

The Oseberg training simulator is a full-scope simulator of the Oseberg A oil production platform located in the North Sea (6). The simulator was originally manufactured for Norsk Hydro in the middle 80's by Norcontrol Simulation and Institute for Energy Technology, to play a central role in the training programme for the operators of the Oseberg A platform.

The Oseberg simulator will in HAMMLAB 2000 be used as “the process” when performing human factors studies related to the oil and gas industry. The Oseberg simulator has been ported to modern hardware and will be modified to handle the recent and future developments in the North Sea, e.g. operation of satellite fields from a centralised control room and remote control of production onshore.

5. Conclusions

Institute for Energy technology is currently performing advanced experimental human factors studies in the Halden Man-Machine Laboratory, mainly within the Halden Project research programme. With the initiation of the HAMMLAB 2000 project the goal is to broaden the scope and domain of studies, by introducing non-nuclear process simulators, first within the petroleum area, and different kinds of nuclear process simulators. In addition, studies related to severe accidents will be possible to perform, due to the extended operational domain of the new simulators.

HAMMLAB will be expanded with new laboratory areas making it possible to perform experiments in parallel, without interfering each other. The laboratories will be flexible with regards to physical size, in order to adapt to small and large studies. A software and hardware infrastructure will be developed, allowing for integration of software systems developed at the Halden Project or in member organisations. Such systems can then be tested in a realistic environment, prior to installation in real-life plants. Based on the above, the Halden Project is confident that the new HAMMLAB will be the global centre of excellence for performing human-machine interaction studies for the management and control of industrial processes, when it is taken into use in the year 2000.
References


Development of a Research Simulator for the Study of Human Factors and Experiments

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Abstract: A research simulator of nuclear power plant for Human Factors was developed. It simulates the behaviors of the 1100MWe BWR nuclear power plant and has almost same functions and scope of the simulation as a full-scale training simulator. Physical models installed in the system enable us to execute experiments with multi-malfunction scenario. A severe accident simulation package replaces the running simulation code when the maximum core temperature exceeds 1200°C and the core approaches meltdown conditions. The central control panel was simulated by soft panels, indicator and operational switches on the panels by computer graphics, displayed on 22 console boxes containing CRT. The introduction of soft panels and EWSs connected with LAN accomplished flexibility and extenderibility. Some experiments by using the simulator were executed and the system has been improved based on the experience from the experiments. It is important to evaluate the effectiveness of any new system by using an actual plant size research simulator before its practical application to keep steady and safe operation of nuclear power plants.

Keyword: human factor, simulator, soft panel, interface

I. INTRODUCTION

It is important to consider not only engineering but also human factors in nuclear power generation to keep steady and safe operations. In designing HMI (Human-Machine Interface), making operation manuals, or developing training methods, must be done based on cognitive and behavioral characteristics, physical limitations, as well as other human factors.

We have conducted Human Factors research in operation by using training simulators since 1984 [1]. Training simulators are good tools for Human Factors research, but we were limited in executing research work by using them [2], for example, it was difficult to change the configuration of control switches, indicators, and alarms to study HMI, to change the simulator program in the scope of automation. Another problem was having the time to use the simulators as much as we needed to conduct research of effects on individuals by shift work because of training schedules. These constraints prevented us from executing some important research items. The reason we faced difficulties in conducting experiments by present simulators is merely that they are designed to be used for training purposes.

It sometimes happens that an idea worked well by using one or two personal computers will not have similar results in a large size system. Nuclear power generation systems are huge, because of this it is necessary to evaluate a system developed by a large-size research simulator before applying the ideas derived from experiments carried out in an experimental room to the actual work places. To accomplish satisfactory research work on human factors, it was desirable to construct a research simulator mainly for the study of human factors.

Our group developed a research simulator for the study of human factors in March 1995 and since then has executed some experiments [3, 4]. This paper describes items discussed before the design, its specification, system's configuration and reports knowledge from experiments by using the system.
II. REQUIREMENTS FOR THE HUMAN FACTORS RESEARCH SIMULATOR

A. Research simulators at present

It is recognized in nuclear power generation field that human factors are the key elements to accomplishing safe operations. After the TMI-2 and the Chernobyl accidents, Japanese people has been interested in human factors of nuclear power generation. Many researchers are conducting studies on human factors in operation, maintenance, design, and other fields of study.

Research simulators for human factors may be classified into two groups: First is the research simulator for a single purpose to develop a certain control room which will be built in the near future. Secondly, one is used for a multi-purpose study of human factor problems by using a simulator based on an existing control room. The former is S3C for development of new N4 in EdF in France. KAERI in Korea has installed a research simulator for designing a new control panel considering human factors [5]. The latter is HAMMLAB simulator for HMI in Halden [6], the Institute of Human Factors in NUPEC has a PWR research simulator for human factors. In the Human Factors Laboratory in JAERI, a Human Factors environment based on nuclear power ship 'Mutsu' was created [7].

B. Specification of the research simulator for Human Factors

What specifications must a research simulator for Human Factors, as a tool, satisfy to execute experiments efficiently? To extract requirements to the research simulator that we were going to build, we collected the subjects of the research based on experience from the research by using training simulators. We then tried to make experimental designs for each item in the table. Items satisfied by the research simulator were classified. Table 1 shows the examples of research items expected to be executed in the future.

The specifications derived from our research designs are as follows:

1) **High fidelity of a reference power generation system**: Research subjects are human, because of this, we will conduct experiments with actual plant operators engaged in everyday service in the final phase of the research work. We want settings as real as possible. For example, if the actual plant has two loops, the simulator must be able to recreate this environment.

2) **High flexibility**: To execute multi-purpose experiments, it must be easy to change freely the arrangement of instruments, switches, indicators, etc.

3) **High extensibility**: New systems such as an operation support system or a large screen display system, must be added to the simulator easily.

4) **Recording of stimulus and response data required**: It is important in human factors research to monitor stimulus and subjects' response data, and to find a relationship between them. The simulator must have the ability to record required data concerning human behavior such as response time, operation, or communications exchanged during their troubleshooting of simulated malfunctions.

5) **Changeability of simulation modules for future use**: The simulation part of this system is a stimulus generation module in light of human factor experiments. The developed system must satisfy usage of the different system if that part is changed from a nuclear power generation simulation to another system. If accomplished, we can extend the scope of the human factors research.

![Table 1: Examples of research items.](image-url)
III. AN OUTLINE OF THE DEVELOPED RESEARCH SIMULATOR

In order to meet the specifications mentioned above, we have developed a research simulator with the following specifications:

1) General purpose EWS: The most important purpose of the training simulator is to behave like the actual reference plant as much as possible. Most of the present training simulators have adopted process computers because they mimic the same performance of the actual plants. It is, however, difficult to change programs in a process computer. The research simulator we have developed has adopted a general purpose EWS because the latest EWS has high performance and there are many good tools available in the UNIX™ operating system for research and human factors data analysis.

2) 22 CRT: Training simulators have almost the same switches, indicators, and recorders as the actual ones to provide trainees the same atmosphere as in the actual control room. We did not adopt the hard switches, indicators, or recorders because we thought that flexibility is more important than the simulation of a real control room. We are focusing on the operator's cognitive process rather than his movement. When the focus of research is on the feeling of operational switches, a training simulator will be used for that purpose. With the introduction of a new method, a soft panel was adopted which was displayed by many CRTs. CRTs can be changed according to the purpose at hand. Configuration of the ABWR new control board is also possible. Furthermore, the configuration of the CRTs can be changed in order to evaluate future generation type control boards.

3) LAN: Ethernet provides us high extendibility, so we can add a new system such as an operator support system or a large screen display system to the simulator easily.

4) Soft panel: No hard switches, indicators and recorders have been made except the full core display part. They are prepared as graphical images made by a UNIX™ application software "DataView™". We have the same arrangement of instruments as the actual control panel in the restricted area. Instruments such as switches and indicators are registered as objects by DataView, because of this, we can easily change their shapes, colors, arrangements, as well as other attribution. Soft panels enable us to execute experiments efficiently. The reference control panel of the soft panels is the second generation type control panel.

5) Physical models to simulate a reference plant: The existing training simulators are made to satisfy the training menu determined first, so the behaviors of a training simulator are good for each training scenario. Multi-purpose research works are also expected in the study of Human Factors. The simulator must satisfy multi-failure scenarios in the experiments. Physical models are needed to simulate a reference plant as much as possible. The research simulator has adopted a quasi-three dimensional model in the core model, SMABRE[8] for part of a heat-water force. We are going to conduct experiments by using simulated multi-failures, because of this, physical model processing is necessary to accomplish high fidelity.

6) Severe accident code: A severe accident code is introduced into the developed simulator. This enables us to study a severe accident.

7) Simulation of the actual system as close as possible: The scope of the simulations is made as wide as possible. The reason for this is to be able to create many different situations. For example, in case that the actual system has two loops, the developed simulator has the same number of the loops.

8) EPG (Emergency Procedure Guidelines) training use: There are two kinds of operation manuals in the control room. One is called an event based manual, the other is called a symptom based manual. The former is good for single, small troubles, the latter is for multi-failures. Aiming for safer operation, engineers are trying to develop hardware reliability of the generation system and operators are requested to have the ability to operate their plant in difficult situations. The research simulator will be used to investigate operator's cognitive process and to develop training methods for operators. EPG training can be executed by using the simulator. Additionally, EPG training use is one measurement for the simulator performance.
9) **General high level computer languages**: Computer programs have been written in FORTRAN and C which enable us to change logic and to add new modules to original ones. Maintenance of the programs is also easy.

10) **The developed system works with data recording systems**: It is necessary in human factors research to collect and analyze operators' behavioral data by VCRs and voice recorders. In our experience of using training simulators, it takes a great deal of time to analyze data relating to human factors. In the developed system, simulator parameters during a simulation session are stored in the EWS and a VCR control computer connected to the research simulator. Therefore, we can easily replay or track back the recorded VCR tapes. For example, if we want to go back to a certain time and replay from that point, we can merely enter the command into the simulator. The VCR tapes would be rewound to the requested time, synchronized with the simulator and would start again with the simulator. An on-line data analysis supporting system is added to the simulator. The system has some rules and extracts error candidates from subjects' operations during the experiment.

**IV. HARDWARE AND SOFTWARE OF THE RESEARCH SIMULATOR**

First of all, we will explain the outline of the system and then, each system in order.

**A. System configuration**

As Fig.1 shows the configuration of the system, the system consists of sub-systems explained below.
1) Interaction System (Photo.1) : The interaction system serves as the interface between the operators and the simulator where operators use the control switches. The system consists of twenty-two CRTs with touch screens, which display simulated control panel and 16 EWSs which control the CRTs (In the beginning of the design, this system consisted of 11 EWS, but later 5 EWSs were added for the improvement of response time).

2) Control System (Photo.2) : The system, consisting of one EWS, sets the simulator in the start-up and shut-down conditions and generates malfunction in the simulated power generation system.

3) Process Computer System : The system processes the computer program which simulates the nuclear power generation system. This system consists of one high performance EWS.

4) Other Systems : Other systems, that are not directly related to the behavior of the simulator, are connected to the research simulator. They are the development system of new CRT pictures for interface, data collection system, and a data analysis supporting system to execute experiments efficiently.

B. Interaction System

1) Soft panel : Soft panel is a panel in which the placement and dimensions of control switches, indicators, and other equipment found on actual control panels in power plants may be entered into a computer. With layout of this panel then being reproduced as a computer-graphics image spread out over a series of 21-inch CRT screens set into a horizontal and vertical array. This design offers an advantage over conventional mechanical simulation control panels in that it is easier to change the shape or placement of switches or gauges on screen. Fig.2 shows an example of soft panels of the second generation central control panel.

This Soft panel is useful for performing tests in which one wishes to judge differences of quality in the ease with which a control panel may be used and monitored occurring as a result of differences in shape or placement of the switches and indicators located on a control panel. More specifically this simulator allows users to create a variety of different types of control panels each with their own layout or placement of switches and indicators. This in turn makes it easy to perform experiments which allow one to find out what differences exist in terms of ease of monitoring or using when given operation are performed with one type of panel in contrast to times when those same operations are performed with another type of panel.
2) Console Box: As we have already noted above, in developing this research simulator for Human Factors, we adopted a design in which a number of CRT monitors are stacked together in a horizontal and vertical array to create a Soft panel capable of providing the simulator with what we consider to be flexibility. There are total of 22 CRTs. To fix them into place each CRT was inserted into a separate console box and these console boxes were then stacked and fitted together (Photo.3). This design allows individual CRTs to be added or removed from the array, and the console boxes can be arranged to represent control panels of a variety of different heights and layouts (Photo.4). Of these 22 CRTs, six of them are designed to be used as desk consoles, i.e., as consoles that can be fixed into place with their screen facing upward.

3) Navigation: Because of the limited space and money for installation, the number of CRTs in the simulator is 22 so that the simulator cannot displayed the whole control panel in the actual control room at one time. We adopted the method that several parts of the control panel are displayed. In the actual control room, operators move to the places where indicators that they want to read are installed and then get some parameters to operate the plant. In our developed system, the soft panel pictures are scrolled instead of operators’ movement (Fig.3). CRT pictures are scrolled by touching on the icons in the bottom on the CRT display.

4) Types of graphical pictures: The developed research simulator has several types of pictures excluding soft panels, as follows;

- CRT picture: CRT pictures are installed in the actual central control board (Fig.4); about 140 pictures are stored in the system.
- Process picture: the pictures based on P&ID show the connections between motors, pumps, and valves (Fig.5); About 90 pictures are stored.
- Educational picture: Because the simulator is made
of a computer program, it has data such as pressure, temperatures in the places where people would not be able to measure in the actual plant. If these data are visualized, operators can gain a better understanding on the relationship between operations and results. For example, fuel temperature distribution, neutron flux distribution etc. About 30 pictures are installed in the simulator (Fig. 6).

- C&L picture: about 5,000 pictures are stored (Fig. 7).
- Logic diagram: Interlocks in the plant are visualized, about 25 pictures (Fig. 8).

C. Control system

Examples of the functions of the simulator are as follows,

1) Backtrack: The function Backtrack enables the instructor to freeze the simulation and load a point of operation for a past point in time during the simulation session. The simulation can then be continued from that point. This makes it possible for example, to study what will happen at different maneuvers starting from a special point of operation. It is also possible to correct mistakes that were made during the first simulation sequence.

2) Malfunction: With the function Malfunction it is possible to simulate various errors and misbehaviors in the technical system, like leakage, pump trips etc. A malfunction specifies actions to take place when some conditions are fulfilled. The instructor defines both the actions and the conditions when building the malfunction. The malfunction can then be activated during the simulation whenever the instructor prefers.

3) Replay: The function Replay, closely related to Backtrack, provides the possibility to replay a simulation sequence. The operator intervening will then be repeated on the operator's system.

4) Curve: With the function Curve it is possible to follow the value of every object, in curve form, that is defined in the process data base. This function is primary intended for use during tests of the process software.

5) Panel Object: Some of the objects in the panel pictures may not be simulated. With Panel Object the instructor can check which objects this applies to.

D. Process Computer System[9]

The scope of simulation is plant specific and covers all main systems and most auxiliary systems. The simulator models consists of modules built up of physical equations. The simulator is designed to simulate the full range of operation in real time, from hot full power to cold shutdown at normal operations as well as accident and system malfunction conditions. The sub-division of the mathematical models follows as far as possible the modular division used for the subdivision of the power plant. Each model is complete in itself with well defined boundaries. The system is tuned to Kashiwazaki-kariwa nuclear power generation site No.5 generation system in TEPCO.

1) The Core Model: The reactor core model is a quasi three-dimensional dynamic model, comprising a
one-dimensional average coolant channel model, with 25 axial zones, together with 43 loosely coupled local channels representing specific channels in the core. The channels selected are the ones containing LPRMs. This gives a three-dimensional representation of the core neutronics with more than one thousand nodes.

The axial variation in the average channel as well as in each local channel can be displayed for variables such as neutron flux, fuel temperature, coolant temperature, void, iodine and xenon concentration. The control rod maneuver groups are all individually simulated and their positions are displayed in various ways.

2) The Thermohydraulic Model (SMABRE) : The SMABRE code is an integrated part of the process model and gives a detailed numerical solution with possibility for LOCA simulation in the range of 0-200% (200% = Guillotine break). The code contains five conservation equations for steam and liquid mass, mixture momentum, steam and liquid energy, using the drift-flux model for phase separation. Other features are hydrogen generation by metal-water reaction and H\textsubscript{2}/O\textsubscript{2} generation by radiolysis. SMABRE has been assessed and verified with LOFT and LOBI.

3) The Containment Model (SCOVE[10]) : The thermohydraulics model is based on multi-component, two-phase equations with thermal non-equilibrium. The code includes individual modeling of H\textsubscript{2}, O\textsubscript{2} and N\textsubscript{2} giving seven conservation equations per compartment. The same set of equations is used both for LOCA conditions and for ordinary conditions and solved in each timestep. Assessment includes comparisons with the Marviken full scale blowdown tests[11] and with internation standard test cases for containment codes.

4) The Core Melt Model (MAAP) : MAAP (Modular Accident Analysis Program) is a severe accident simulation package. The code replaces SMABRE and SCOVE when the maximum core temperature exceeds 1200°C and the core approaches meltdown conditions. MAAP models a wide spectrum of phenomena including injection of water from several sources, the blowdown of water, steam and hydrogen from the nuclear boiler. The most important parts of the code are: modeling of core uncover and heat up, cladding oxidation, hydrogen generation and transport within the reactor vessel and containment, fuel melting and movement, reactor vessel melt-down, melted core (corium) interaction with drywell floor concrete, and fission product release, transport and deposition.

5) The Control and Logic Management Tool (ASK) : ASK (Automatic System Configuration) is a symbol-oriented tool for construction, documentation and simulation of control and logic. It uses a relational database (ORACLE\textsuperscript{TM}) connected to a graphical interface (AutoCAD\textsuperscript{TM}). The system executes the control and logic diagrams once they have been drawn, without additional programming. The control and logic diagrams are integrated with the simulator during simulation. It is thereby possible to follow the dynamic behavior of the interlocks during simulation by watching the response in the drawings. At the same time the process response and the panel response may be followed on other pictures.

Photo 5 Data collection system.

Photo 6 Data analysis supporting system.
V. RESEARCH EXAMPLES AND DIFFICULTIES

We will explain the research works that we have conducted by using the research simulator and difficulties we faced in the works.

A. Research examples

1) Study of Possible Effects of the Change to the SI Unit on Operators’ Responsive Actions[12]: By the execution of new measurement system, a unit system used in nuclear power plants will be changed from the MKS system to the SI system (Système International d’Unités). The purpose of the research was to investigate the effect of the change and to discuss countermeasures against the effect before installation of new unit system into the actual power plants.

Indicators, recorders, and CRT pictures converted to SI units were made by soft panels. Numbers written in the operation manuals were converted to a new unit system and rewritten. Actual plant operators worked against simulated troubles by using the converted soft panels. Their behavioral and communication data were analyzed. Questionnaires and interviews were also executed after the experiments. Based on these data, we discussed the possible effects of change to SI units. Much effort was expected to change indicators, recorders, CRT pictures on the control board into a new unit system, but actually it was easy to change them because they were made of computer graphics (Fig.9). This experiment was one example that showed the flexibility effec-

![Photo 7 ABWR type central control panel.](image)
tiveness of interface parts of the research simulator.

2) Improvement of annunciation system in the central control panel: In the experiments by using training simulators projected by BWR group [13], the improvement of annunciation on the control panel was expected to make better operators' performance such as the response time from indication to the trouble, or detection of another annunciation after an occurrence of a failure. There are about one thousand pieces of annunciation tiles on the central control panel. Only one frequency of the annunciation sound is installed in the first and second generation type control panel. To support operators' understanding of the annunciation, we made several annunciation sounds and carried out experiments to discuss annunciation systems. Repetitive sounds of one frequency used in the present control panel, musical sound, voice alarm, and their combination were presented as stimulus and evaluated from the viewpoint of effectiveness to operators' performance.

In this experiment, the sound generator had to run with the simulator synchronizing. It is difficult for the training simulators to connect another devices to them, on the other hand, it was not difficult for the research simulator to connect new devices because the simulator has TCP/IP in communication with other computers. This experiment also shows the effectiveness of extendibility from the research simulator.

3) A Study of large display panel installation in the existing control panel: In the final Phase IV of the research work by using training simulators conducted for twelve years done by Japanese BWR group, our research interest was HMI. We analyzed the effect of interface which contributed to plant operators' performance by using ABWR type central control panel simulator (Photo.6). Results showed that the large display panel was quite effective for plant operation [14]. Therefore, we started studying how to insert this kind of large display to the existing control panel. In the existing central control room, certain size limitation of large display panels can be installed because of space limitation. We are discussing about what information must be displayed in the limited size. We are going to execute experiments by installing a large display panel to the research simulator and collect operators' behavioral data and their subjective evaluation after operation under multi-malfunctions.

4) CRT picture guidelines based on the characteristics of human cognition: It is expected that a flexible display / operation system like a CRT or liquid crystal display will be adopted as a HMI device in the future. Unfortunately, there is few study reports for practical application on information display and navigation methods for practical use in the actual plant. We, therefore, are investigating the design method of CRT display usage based on the characteristic of human cognition by executing Human Engineering-based basic and applicational experiments.

5) An effective education and training method for NPP operators: The most important role of NPP operators required high level safety is the ability not only to conduct skill-based operations but also to take countermeasures based on knowledge about their plant against an accident. We are developing education and training materials for operators that enhance their knowledge.

Fig.10 Graphic display of MAAP.

Fig.11 Graphic display of core condition.
level by using the research simulator. The simulator gives computer graphic of results from MAAP and the internal condition of the core to trainees for an easier understanding their operations and results (Fig.10).

B. Feedback from experiments using the research simulator

The problems we faced in experiments using the research simulator will be explained then countermeasures against them are as follows,

1) Response time of the soft panel: In the first version of the simulator, response time from the soft panels was known a little bit later than we had expected, and this brought some difficulty in the execution of experiments. We, therefore, added another five high powered EWSs to the simulator, sixteen EWSs can run twenty - two CRTs. This revision solved the problem of response time.

2) Presentation of annunciators: The soft panels are displayed by twenty-two CRTs, which are rather small in comparison with the actual panels, so it is impossible for operators to see the whole panel at one time. The problem was the detection of an annunciator that appeared in the undisplayed parts. Of course, this problem would be expected before an experiment. We had made special icons that would show the appearance of annunciator in other panel, but these icons were not so effective to inform operators of the problem. We added other icons for revision and newer ones have proved more effective (Fig.12).

3) Connection of experimental devices to the simulator: The system consists of computers connected with LAN. It was easy to add an experimental device to the system with hardware connection and protocol by TCP/IP. Some amount of efforts was needed to adjust the data transferring format such as data type, cycle, unit, and other information between the simulator and devices. We are planning to prepare a common format in the simulator for easy exchange of data between systems to reduce these efforts.

4) On-line data analysis supporting system: There are several hundreds of items in the simulator. It took time to find the items for making analysis logic. The retrieve function was added to the system.

5) VCR control system and execution of experiments: It took time to find the start mark in the VCR tape after the simulator had started. This timing delay affected the experiment execution. We will change this analogue recording device to a new digital video recorder in the future.

6) Learning effect of operators: It takes time to operate the simulator by using soft panels. Before execution of experiments operators received a one-day training to accustom themselves with the new interface. The three-day training time would satisfy their knowledge of this interface.

VI. CONCLUSION

When we installed the research simulator in the newly constructed technology development center in March 1995, there were few simulators similar to the developed one. There was one in HAMMLAB for HMI research which matched its development. In a few years, some research facilities have created research simulators. The circumstance for human factors research became rapidly well prepared.

Our research simulator is the first one for multipurpose Human Factors research works, in the nuclear field, in the world except for the research simulator for HMI in Halden. A simulator is a necessary tool for executing Human Factors research. As mentioned before, it sometimes happens that the application of results received in the small experimental laboratory, to the actual large system, would not work well. For example, we faced the same problem in the process of developing
our research simulator. The new idea of soft panels has no problem if the system is small and the response time is satisfactory and fast. In the case of the developed research simulator, the number of variables increased was huge, it took time to transfer data from the process computer system to the interaction system. The nuclear power generation system is large, because of this it is necessary to make the system as large as the reference power plant and to evaluate it by experiments before actual application. The developed research simulator is rather large if compared to an engineering simulator. We will execute Human Factors research work by using the developed simulator to aim for safer operation of nuclear power plants[15].

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AN OPERATOR SELF-TRAINING SYSTEM BASED UPON THE EMULATION OF INSTRUCTOR SKILL

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ABSTRACT

In the early 90's a project started in Ansaldo for developing the Training Compact Simulation Technology. One of the objectives was to substitute the human instructor with an Expert System based on the Emulation of Instructor Skill (EIS). EIS stems from software architecture and an ad hoc language (ABIS) developed at Institute of Control Sciences (ICS) of Moscow. The expert system relies upon Artificial Intelligence technology and qualitative logic modelling of controlled plant physics. Its task is to explain the physical effects of controlled plant behaviour and control system response and comment on them as well as estimation of operator's actions in standard and non-standard situations. Insight into process physics is obtained through consulting the expert system into which the qualitative process model is built-in, thus enabling expert explanation of fine physical effects such as: identification of disturbance and control system actions.

The main features of EIS are: a) checking the correct memorisation of the plant procedures, b) testing his own reactions when facing abnormal and/or unexpected situations, c) improving his own engineering sensitivity, d) getting on-line answers to specific questions, such as: what are the reasons of the current status of Plant (Operator control actions, control system actions, disturbances imposed by the Instructor, transient behaviour of parameters); what events are the result of the action performed by the Operator (step by step); what events can result from a given Operator action (prediction); correspondence of the Operator actions to the steps of the performed procedures; in abnormal situations, judgement about operator actions (positive or negative, with respect to the goal of plant stabilisation or disturbances elimination).

A demonstrative application was developed for Sampierdarena 40 MW Cogeneration Plant. The plant model running under the simulator was built-up by means of the LEGO code. The LEGO code is a modular package developed at the Research and Development Department of the Italian National Electricity Board (CRA-ENEL) to facilitate modelling of the dynamics of fossil-fuelled and nuclear power plants. The
LEGO code consists of a library of pre-programmed, pre-tested and pre-validated modules, that represent power plant components and a master program which allows the user to build-up a model by automatically interconnecting the modules in the arrangement determined by the modeler.

1. The ABIS language

The ABIS language has been developed to implement deduction systems with databases dwelling upon extended relational data models. This accounts for the set of basic language elements and their features. According to ABIS, a deduction system consists of a knowledge base (KB) and a database (DB). The KB consists of a set of rules of the following form:

If: <condition 1>, ..., <condition n>
Then: (consequence 1>, ..., <consequence n); confidence

The DB consists of a set of facts with assigned confidence

<fact 1, confidence 1>, ..., <fact k, confidence k>

The confidence is used as an additional component of both rules and facts. It characterises the degree of confidence in the rule (or fact) under particular current conditions in the KB (or DB), and is represented numerically by a value ranging from 0 to 1. The basic process in the deduction system (called inference cycle) is made up of:
- the comparison of the facts available in the DB with some specimens or comparison patterns contained in the conditions of KB rules;
- identification of the conditions under which correspondence to the patterns takes place;
- derivation on their basis of appropriate conclusion in the rule consequences, thus updating DB.

This inference cycle is repeated until DB stops changing.

2. The qualitative modelling

The models of controlled plants describing the state of the equipment, the process parameters and their interrelations by means of qualitative scales, in contrast to the traditional numerical approaches, are identified as qualitative process models (QM). The approach relies upon the following simple idea: when analysing and forecasting the controlled plant state, a skilled operator solves no differential equations, but tries to model qualitatively interactions and propagation of disturbances, their passage through connected subsystems, their cause-effect relations. It is precisely this goal that is set also before the qualitative modelling approach.

The construction of a deductive system is one of most popular ways of designing a
qualitative model. The applications of QM to the problems underlying self-training simulator are discussed in more detail. The tasks of a QM are the following:
- detection of cause-effect relations between real events in the controlled plant;
- detection, among these events, of the initiating ones, i.e. those caused by the disturbance source, rather than being secondary, i.e. due to the disturbance propagation;
- detection of the causes of these events such as:
  a) erroneous operator actions;
  b) control system defects;
  c) process equipment defects.

![Diagram of QM process]

**Figure 1**

The concept of initial event plays a pivotal role in the QM technology involved in the self-training system. The events can be of the following type:
- all operator actions upon process equipment (pump on/off, manual change of actuator position, on/off of isolation valves, etc.);
- discrete actions of the automatic control system (operation of process algorithms, protections, blockings, etc.);
- disturbances caused by the equipment defects detected at the previous diagnostic stages (in the simulator these are the disturbances introduced by the Expert System).

Each initial event (see fig. 1) is modelled independently in real-time by the QM facilities for the plant state that took place at the instant of the event occurrence. As a result of each initial event j, there appears a route j of possible qualitative deviations of the process parameters from their normal behaviour.

Next, all the initial event routes are superimposed in time, and the resulting picture is compared continuously to the events in the physical plant (actually in the simulator). This comparison enables elimination of minor deviations as predicted by the QM and to answer the question of what events and by what routes have given rise to one or another event in the physical plant.

This brings up the problem of designing mechanisms for determination of the routes of possible deviations, or, to state it in a different way, how and in what terms describe the propagation of the initial local disturbances through the equipment and process subsystems.

The process of qualitative model design involves the following stages:
1. determination of the composition of standard plant components as exemplified by pipelines, valves, pumps, heat exchangers, etc., for power process;
2. choosing qualitative parameters describing component behaviour and their interrelations; introduction of qualitative scales for measuring these parameters; for example, the following scales can be used for measuring the deviations of process parameters such as pressure, temperature, engine speed, etc.: [below norm, norm, above norm], [rapid growth, growth, ... ];
3. description of component operation rules, that is interrelation between the input and output qualitative parameters of a component;
4. description of component topology within subsystems;
5. identification of weakly related subsystems with the aim of decomposing the qualitative model and supporting its real-time operation; choice of the parameters describing the interactions between subsystems;
6. description of subsystems interaction rules and interactions.

The following problems arise at practical implementation of the qualitative model:
- how to describe equipment topology;
- how to describe equipment hierarchy and system processes;
- how to decompose the controlled plant into subsystems;
- how to take into account superposition of influences of different disturbance sources on the same parameter;
- how to describe subsystem interactions;
- how to support, real-time operation;
- how to take into account the temporal relations between the system events.

The main mechanism for implementation of a QM is a deductive system. It has a KB and a DB. The KB has a set of rules of the following form:

\[
\text{If: } \langle \text{condition } 1 \rangle, \ldots, \langle \text{condition } n \rangle \\
\text{Then: } \langle \text{consequence } 1 \rangle, \ldots, \langle \text{consequence } n \rangle;
\]

The DB consists of a set of facts like these:

\[
\langle \text{fact } 1 \rangle, \ldots, \langle \text{fact } k \rangle
\]

No confidence degree is required in this context, because rules and facts are taken from real experience and real observation. As in the general case, the inference cycle compares the facts included in the DB with the conditional parts of the rules included in the KB, thus determining their corresponding consequences as new facts to be added to the DB. The inference cycle is repeated until no new fact is generated and added to the DB.

Some inherent features of ABIS language allow to answer the practical problems listed above. The key element is the extended relational data model whose basic notions are "attribute", "domain" and "tuple". As an example of a fact (i.e. an element of the DB), let us consider the following tuple describing a gate type valve with name LC10S located in pipeline section LC10Z of subsystem LC that is closed by operator 90 s after the time origin:

\[
\text{valve(Index = LC10S, Module = LC, Type = gate, Section = LC10Z, Condition = 0, } \\
\text{Norm = 0, Initiator = oper, Time = 90, Status = ex)};
\]

The behaviour of each plant component is modelled by a group of rules called a module. An entire library of such modules has been developed with the support of process plant experts. As an example, one of the rules included in the pump module is reported herebelow, translated into descriptive language:

\[
\text{If: } \langle \text{discharge section of Pump A is Y1} \rangle \ \text{AND} \\
\langle \text{flow (dynamics) on this section is } _P \rangle \ \text{AND} \\
\langle \text{suction section of Pump A is Y2} \rangle \\
\text{Then: } \langle \text{flow (dynamics) on section Y2 is } _P \rangle;
\]

3. LEGO Code Overview

The LEGO code is a modular package developed at the Research and Development Department of the Italian National Electricity Board (CRA-ENEL) to facilitate modelling of the dynamics of fossil-fuelled and nuclear power plants. The LEGO code consists of a library of pre-programmed, pre-tested and pre-validated modules, that represent power plant components and a master program which allows the user to build-up a model by automatically interconnecting the modules in the arrangement determined by the
modeler. Each module describes a physical plant component to the prescribed level of fidelity and is independent of any other module. A module consists of a lumped parameter model, derived from first principles, describing a physical process by means of a system of non-linear algebraic and/or differential equations. A single component can be represented by different modules of different level of complexity to meet different modelling needs.

The basic characteristics of the LEGO package, can be summarised as follows:
- modularity: component models (modules) are available for general plant components (such as pipes, valves, pumps, heat exchangers, tanks, etc.) and the user can connect them in accordance with a specific plant design;
- flexibility: the user can solve special modelling problems by developing the mathematical model of special components, which can be included in the module library of the package;
- reliability: all the numerical algorithms used by the package are centralised in the master program. Module notifications, due to different mathematical modelling assumptions, do not require any numerical algorithm updating.

Moreover the modules can exchange information among themselves only by means of the master program so that they can be considered independent.

With respect to the numerical problem the main features of LEGO are:
- simultaneous solution of all non-linear algebraic and differential equations, using an implicit formula for the numerical integration method.
- use of sparse matrix techniques in order to reduce computation time, dealing with large power plant models.
- steady-state computation, allowing interchange of the role of input variables, output variables and uncertain constant parameters.

4. Sampierdarena Cogeneration Power Plant Simulator

The Sampierdarena power station is a combined cycle plant dedicated to produce both electrical and heat power. The thermal power is sent to final users as superheated water. The main plant components are: 1 gas turbine rated 21 MW, 1 steam generator where the residual heat of turbine exhaust gases is recovered, 1 steam turbine rated 9 MW, 1 generator rated 37.5 MWe, 1 583 m³ steam condenser, 1 deaerator.

The gas turbine, the electric power generator and the steam turbine are placed on the ground mounted on the same axis as a unique group.

The reference plant has been modelled by means of LEGO code modelling tools. The resulting simulation model is made up of 8 submodels: four of them reproduce process subsystems and the other ones reproduce automation system. More than 2000 variables are handled.
5. The main features of the self-training system

The concept of substituting the human instructor with the Emulation of Instructor Skill has been implemented by connecting the simulator with an Expert System which includes the Instructor Skill basically in a qualitative model of the plant. When connected to the real-time simulator, the Expert System allows the Operator of the combined cycle cogeneration plant to make self-training, also in absence of the dedicated Instructor. Particularly, a session of automatic training managed by the ES enables the Operator to:

a) train himself in the correct execution of the plant procedures;
b) verify his own reactions when facing abnormal and/or unexpected situations;
c) improve his engineering sensitivity.

Through the graphic interface of the system the Operator can control the simulator (run, freeze, stop) and communicate with the ES in order to:

1) Monitoring the current status of the plant and the transient behaviour of the parameters.
2) Individuate the causes of the eventual parameters deviations:
   - Operator control actions
   - control system actions
   - disturbances imposed by the “Instructor”, i.e. by the ES
3) Recognise the happened events as consequences of the Operator actions (step by step)
4) Predict the consequences of a given Operator action, in terms of deviation of plant parameters from normal values (what-if)
5) Verify the correspondence of the Operator actions to the step of the performed procedures.
6) In abnormal situations, have a judgement about the performed actions (positive or negative, with respect to the goal of plant stabilisation or disturbances elimination).

During a training session, the Operator can verify his ability in recognising abnormal plant situations. This is the reason why he can set a difficulty class (from level 0 to level 4) at the beginning of the session. Thus the ES will introduce random disturbances of an increasing difficulty.

6. The built-in functions of the self-training system

The simulator connected to the ES is intended also for independent training of process plant Operators and for teaching them the skills of control in normal and emergency situations. Use of the simulator requires knowledge of the fundamentals of physics, heat
transfer engineering, technology and control at the level of technical schools. To attain the targets of training that are hierarchically ordered in terms of knowledge complexity and quality, the simulator may be used in various operational modes such as:
1) training in the fundamentals of controlling a particular process plant;
2) training in automatic execution of standard instructions in standard situations;
3) getting an insight into process physics;
4) training in non-standard situations requiring profound technological knowledge of a particular controlled plant and consummate skills in operative diagnostic and decision making.

The self-training simulator has two major parts: the plant operator workstation and the expert system. The simulator dwells upon a numerical process model and presents all the possibilities of monitoring and control in an environment close at most to the physical control system.

The expert system relies upon the AI technology and qualitative logic modelling of controlled plant physics. Its task is to explain the physical effects of controlled plant behaviour and control system response and comment on them as well as estimation of operator's actions in standard and non-standard situations.

The basic functions of the operator workstation include:

a) monitoring the state of the simulator;
b) displaying the state of equipment and the values of the process parameters;
c) displaying the messages of the control system;
d) performing the following control actions:
   - equipment on/off;
   - flow rerouting;
   - controller setting up;
e) monitoring of process parameter trends.

The automatic execution of standard procedures is trained by the instructor expert system that performs the following functions:
- monitoring step-by-step the correct execution of the instructions;
- generating general estimate on the base of operator action adequacy to the procedures.

This operating mode is started by choosing a procedure that the operator wants to study. Next, the numerical model is set up to the initial state from which the desired procedure is to be executed. Time monitoring is switched on and the trainee proceeds to execute procedure steps in compliance with the operator manual. For convenience, manual contents is displayed in a screen window.

Step-by-step monitoring of procedure correctness is done by automatic monitoring of numerical model state and comparison of operator's actions with the correct ones as defined by the manual.
From procedure execution results, a combined report on the trainee is compiled containing, in particular:
- protocol of operator actions with time marks;
- estimation of operator's actions in terms of the operation manual.

Insight into process physics is obtained through consulting the expert system into which the qualitative process model is built in, thus enabling expert explanation of fine physical effects, in particular, such as:
- identification of disturbance propagation paths along the system;
- detection and demonstration of cause-effect relations between events in the controlled plant;
- estimation of operator action advisability in an arbitrary situation;
- prediction of the effects of operator actions, external disturbances and control system actions.

Operation of the qualitative and quantitative models are timed. The operator can suspend operation in any mode, consult ES and, then, resume his (her) work.

ES operates in the question-answer mode: by means of the dialogue facilities the trainee formulate requests about an event or phenomenon and gets an answer in terms of text messages and technological graphics.

ES consultations are as follows. First, the trainee chooses a former or current event such as:
- operator action;
- operation of protections or blocking; or simulated equipment failures.

As long as no other event is chosen or the ES session is not completed, the trainee may ask about the chosen event and current process situation. There are four types of questions that can be answered by ES.

The first type of questions to ES enables the trainee to identify the paths of disturbance propagation through the subsystems and get an integrated picture as generalised mnemonic, with disturbance propagation zones shown in different colours, of disturbances caused by the event in individual process subsystems.

The second type of question to ES concerns cause-effect relations between process events and parameter behaviour. In doing so, explanation may pursue both directions:
- for an event chosen by the trainee, ES can indicate affected parameters;
- for an abnormal process parameter, ES can reconstruct the sequence of events resulting in this disturbance.

The third type of information provided by ES is estimation of action advisability in current situation from the viewpoint of operator's global aim of maintaining the basic process parameters within the norm. An event is regarded as advisable if it is oriented to offsetting disturbances and normalisation of the parameters. Here, advisability is
estimated for each individual action and process parameter.

The fourth type of questions to ES makes use of the forecasting capabilities of the qualitative model. ES enables determination of future trends induced by operator actions, external disturbances or failures. If the trainee desires, ES can correlate the past events with future plant behaviour.

For non-standard situations requiring profound knowledge of plant technology and consummate skills in diagnosing situation and making control decisions, training is done by simulating single/multiple failures at random instants. For estimation of trainee's actions in situations that are out of the scope of instructions, the expert system relies on process knowledge and the following global control aims:
- maintaining process productivity;
- maintaining major process parameters within the normal range;
- execution of given instructions for changing the controlled plant mode.

Interaction with the simulator can follow two scenario variants:
- scenarios with failures in stationary states of controlled process;
- scenarios with failures occurring at transitions from one normal process mode to another.

Additionally, the simulated failures are chosen randomly depending on the complexity class:
- LOW (lower class);
- MEDIUM-LOW (medium-lower class);
- MEDIUM-HIGH (medium-high class);
- HIGH (high class).

Session begins with choosing scenario type, if necessary, the procedure to be executed and the complexity class. Relying on these set-ups, the simulator is adjusted to the desired state of the controlled plant, and the training starts.

One or more failures are modelled at random times after session start. The trainee must do fast and correct diagnosis, detect failures, make decision about offsetting them and do appropriate actions.

7. Conclusions

Automation has an important role in the operation of plants. Many processes are monitored and managed by mean of sensors, actuators and logic of the automation software.

The automation process started from the automation of components, extended his functions to the automation of single processes or part of them, and now reached the upper part of the decision making pyramid.

This last part of the automation process can not be completely covered, and is now
characterized by the assistance of operator aids: instructions and procedures and in the best cases simulators used under the supervision of a skilled instructor.

To make some steps further in the automation of plant operation, Ansaldo and ICS made the research presented in this paper, and realized a tool with the final goal of:
- allow the operator to train himself without the need of instructor assistance
- substitute the instructor skill with a Qualitative model of the plant
- substitute the instructor advice with QM predictions of the plant future behaviour.

A simulator without the need to have an instructor assistance, but anyway, with proper guidance, allow a more frequent usage, with less limits, allowing to achieve more knowledge of the process.

The QM realized, similarly to up-to-dated tools, is based on modules that can be personalized to build new applications. The formalism of representation of the knowledge and the inferential mechanism allow partially to justify the conclusions. As a future work can be envisaged to improve the explanation mechanism to help operator to better understand the cause-effect of events. The QM has good performances and makes reasonably good predictions in advance.

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- **FUNCTIONING BY ANALOGY**
Tree Simulation Techniques for Integrated Safety Assessment

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1. Introduction

In the development of a PSA a central role is played by the construction of the event trees that, stemming from each initiating event considered, will finally lead to the calculation of the core damage frequency (CDF). This construction is extensively done by means of expert judgement, performing simulation of the sequences only when some doubt arises (typically for small break LOCAs) using integrated codes. Transient analysis done within the framework of the PSA has important simplifications that affect the transient simulation, specially in relation with the control and protection systems, and may not identify complicated sequences. Specialised simulation tools such as RELAP5 have been used only at very specific points of the PSAs due to time and model complexity constraints.

On the other hand, a large amount of work and expertise has been laid in the development of simulation tools, independently of the PSA effort. Tools simplified in the phenomena they consider, but complete in the treatment of automatic and manual control and protection systems are now available for automatic generation and testing of the sequences appearing in the Event Trees. Additionally, an increasing effort and interest has been spent in the development of efficient means for the representation of the Emergency Operating Procedures, which, incorporated into the simulating tool would make a more realistic picture of the real plant performance.

At present, both types of analyses —transient and probabilistic— can converge to be able to dynamically generate an Event Tree while taking into account the actual performance of the control and protection systems of the plant and the operator actions. Such a
tool would allow for an assessment of the Event Trees, and, if fed with probabilistic data, can provide results of the same nature. The basic requirements for this tool are described in figure 1.

Key aspects of this approach that imply new theoretical as well as practical developments in addition to those traditionally covered by classical simulation techniques are:

a. An unifying theory that should i) establish the relationship among different approaches and, in particular, be able to demonstrate the standard safety assessment approach as a particular case, ii) identify implicit assumptions in present practice and iii) establish a sound scientific reference for an ideal treatment in order to judge the relative importance of implicit and explicit assumptions. In addition, the theoretical developments help to identify the type of applications where the new developments will be a necessary requirement.

b. The capability for simulation of trees. By this we mean the techniques required to be able to efficiently simulate all branches. Historically algorithms able to do this were already implemented in earlier pioneering work for discrete number of branches while stochastic branching requires Monte Carlo techniques [1, 2, 3].

c. The capability to incorporate new types of branching, particularly operator actions

This paper shortly reviews these aspects and justifies in that frame our particular development, denoted here as Integrated Safety Assessment methodology [4]. In this method, the dynamics of the event is followed by transient simulation in tree form, building a Setpoint or Deterministic Dynamic Event Tree (DDET). When a setpoint that should trigger the actuation of a protection is crossed, the tree is opened in branches corresponding to different functioning states of the protection device and each branch followed by the engineering simulator. One of these states is the nominal state, which, in the PSAs, is associated to the success criterion of the system.

Several damage variables can be defined in the ISA methodology, leading to different risk measures. Thus, the analysis does not restrict to the core damage condition, but also to other risk conditions that may arise in the course of the DDET unfolding, including
design criteria. Additionally, the safety assessment will be more accurate and new type of lessons can be learned from it, for instance assessments of the settings of protection systems, as well as of the success criteria and sequence delineation of the PSA model.

The details of the paper show how ISA can be mathematically derived from the Theory of Probabilistic Dynamics remaining compatible with today's state of the art, taking credit of the present efforts and applying their results into a next-step improvement. The practical issues of implementation and specifications of the computational tools needed to obtain a usable tool for the analysis of the safety of nuclear facilities are also discussed.

2. Integrated Safety Assessment Methodology

2.1 Theoretical framework. Theory of Probabilistic Dynamics (TPD)

Considerable research efforts have been undertaken in Dynamic Reliability, as recently reviewed in [5]. In particular, the theory of Probabilistic Dynamics, developed in [6], where the Chapman-Kolmogorov equation is used to find the probability density of state \( \tilde{\mathbf{f}}, \pi_{\tilde{\mathbf{f}}}(\tilde{x}, t) \).

The state of the facility is represented by:

The plant status vector: a set of time-dependent integer numbers \( \tilde{\mathbf{f}}(t) = (j_1(t), \ldots, j_N(t)) \), describing the different states of the facility systems (e.g. nominal, derated, failed in a given mode, etc.).

The process state vector: a set of time dependent process variables that are necessary and sufficient to determine the dynamic evolution of the damage, i.e. the real vector \( \tilde{x}(t) = (x_1(t), \ldots, x_N(t)) \) describing the facility process evolution (temperatures, flows, etc.).

The evolution of the process vector \( \tilde{x} \) given a plant status \( \tilde{\mathbf{f}} \) is described by the equations

\[
\begin{align*}
\dot{\tilde{x}} &= \tilde{f}_j(\tilde{\mathbf{u}}, t) \\
\tilde{x} &= \tilde{g}_j(\tilde{\mathbf{u}}, t)
\end{align*}
\]

(1)

\( \tilde{\mathbf{u}} \) being the process vector at the instant the plant status became \( \tilde{\mathbf{f}} \).

It is also convenient to distinguish between connected states and disconnected states of a system: connected states are those influencing the dynamics of the system, i.e. determining \( f_j \) in equation (1), while disconnected states do not influence the dynamics.
2.2 Brief description of the present methods

Traditional accident simulation obtains the response of a large system with the dynamics corresponding to a facility status \( j \), from initial conditions \( \vec{u} \) and described by equations (1) where \( j \), initially in a steady state before the initiating event occurred, changes at times \( t_n \), with \( x_n = u_{n+1} \) due to the intervention of safeguard systems or protections as a result of exceeding setpoints for automatic actions (or alarm setpoints for manual), constituting the event sequence. The design risk requirements may then be checked by grouping the initiating events according to a probability classification and verifying design safety limits. As already indicated, this approach falls short of conclusions because the events seldom degrade enough and the probability of potential, more degraded sequences, is not calculated.

On the other hand, in the traditional level 1 PSA, the probability is calculated for a set of degraded sequences previously identified by engineering judgement with the help of separate simulation calculations (sequence delineation). The method first finds for a sequence of frontline system states, \( \vec{j}_n \), all possible combinations of the states of their components\(^1\) \( c_1, c_2, \ldots, c_k \) leading to \( \vec{j}_n \) (for instance leading to a faulted state if failure to satisfy success conditions, understood as sufficient requirements to prevent core melt) by mathematically describing an appropriate logic, usually through a fault tree boolean function \( \Phi(\vec{c}) \). Then, the sequence probability is evaluated or maximized by solving a classical Markov system or approximations to it if the basic events are considered independent. Finally, and again by engineering judgement or separate dynamic calculations, the sequences are discriminated to lead or not to core-melt and the overall probability of core melt evaluated. In this case, the evolution of \( \vec{x} \) with time according to equation (1) is taken into account only in the sequence identification stage and in the considerations behind success criteria and core melting sequences, both a crude treatment of the dynamics as will be seen below.

From this brief description it is obvious that in one approach it is the evolution of the process vector \( \vec{x} \) what matters and, in the other, that of the status vector \( \vec{j} \). Obviously then, an integrated treatment of combined states \((\vec{j}, \vec{x})\) taking into account both evolutions simultaneously will conciliate both ways of looking into the problem.

2.3 The ISA approximations

Let us now assume the Markovian states \( \vec{j} \) partitioned in two groups. Let \( \vec{j} = (\vec{\alpha}, \vec{\beta}) \) be the Markovian state of the system where \( \vec{\alpha} \) labels the first group and \( \vec{\beta} \) the second. The second group concerns disconnected states and transitions between themselves, which we call \textit{conditioning events}, while the first involve transitions triggering facility transients, including connections and disconnections as well as events between already connected states. The transitions of the first group include the initiators as well as failures under

\(^1\)\( c_i \) is a vector of integer numbers describing the state of the components of facility systems.
demand conditioned by setpoint signals of the automatic or manual systems that are supposed to occur with deterministic delays after the setpoint is crossed.

We further assume that the duration of the transient or accident (i.e. the time between two process steady states), is short enough for the second type of transitions to have a very low probability during the transient. Because while in steady state the setpoint transitions cannot occur, we are lead into large steady periods of passive events with $\vec{x} = \vec{x}_{\text{steady}}$ followed by short transients that originate dynamic sequences through setpoint transitions. As will be seen, for this important practical case, that covers most real situations of interest with the exception of operator actions (see below), there exist solutions that can be handled with existing computational technologies.

We write therefore

$$
\pi_j(\vec{x}, t) \equiv \pi(\vec{\alpha}, \vec{\beta}, \vec{x}, t) = \pi(\vec{\alpha}, \vec{x}, t | \vec{\beta})\pi(\vec{\beta}, t)
$$

(2)

where $\pi(\vec{\alpha}, \vec{x}, t | \vec{\beta})$ is a conditional probability reflecting the influence of passive events ($\beta$ transitions) on the probabilities of response of systems to setpoint demands, through common basic events and other dependencies, while transitions in the second group are not conditional.

If the current steady state began at time $T_0$ and an accident, i.e. an initiating event, occurs at time $T$ and lasts until $T + \Delta T$, in the preaccident period there is no evolution of the $\vec{x}$ process vector, so that no $\alpha$ transition can happen, other than the initiator itself, then

$$
\pi(\vec{\alpha}, \vec{\beta}, \vec{x}, t) = \pi(\vec{\alpha}, \vec{x}_0, T_0 | \vec{\beta})\pi(\vec{\beta}, t) \quad T_0 < t < T
$$

(3)

This is the case of conditioning events in the time interval in which they are actually occurring. Once the initiating event of the accident happens at time $T$, the evolution during the transient can be described by

$$
\pi(\vec{\alpha}, \vec{\beta}, \vec{x}, t) = \pi(\vec{\alpha}, \vec{x}, t | \vec{\beta})\pi(\vec{\beta}, T) \quad T < t < T + \Delta T
$$

(4)

since on a time scale ($\Delta T$) much shorter than $T$, the only transitions to be examined are those of the first group as determined by the dynamics of $\vec{x}$ and the setpoints.

We have then decoupled the analysis of the conditioning events occurring during steady states periods, which is to be calculated as a classical Markov evolution from the analysis of accident sequence initiators where the equations of probabilistic dynamics have to be solved with a reduced set of transitions. Refer to [7] and references thereof for the details of the above calculations.

Besides, if during the accident period we take into account the short term assumption, not only $\beta$ transitions are prohibited, but also the random $\alpha$ transitions (i.e., no other initiator will occur during the considered transient duration). Under these conditions, the sequence of events occurs only upon setpoint demands. It has been proved [4, 7] that if
all events occur upon setpoints demands, the solution is given by

\[ \pi_j(t) = \pi_0 \cdot \sum_{n=0}^{\infty} \sum_{Path_n} \prod_{l=0}^{n} X_{j_{l+1}}(u^l_{l+1}) \]  

(5)

where \( \pi_0 \) is the initiator frequency, \( \Theta(t) \) is the step function, and each path is given by

\[ Path_n \equiv \begin{cases} \bar{j}_0 \rightarrow \bar{j}_1 \rightarrow \bar{j}_2 \rightarrow \ldots \rightarrow \bar{j}_n \rightarrow \bar{j} \\ \bar{x}_0 \rightarrow \bar{u}^{l_0} \rightarrow \bar{u}^{l_1} \rightarrow \ldots \rightarrow \bar{u}^{l_{n+1}} \rightarrow \bar{x} \end{cases} \]

\[ \tau_n \equiv \tau_n(\bar{x}_0) \equiv \tau_{j_0}(\bar{x}_0 \rightarrow \bar{u}^{l_0}) + \sum_{i=1}^{n} \tau_{j_i}(\bar{u}^{l_i} \rightarrow \bar{u}^{l_{i+1}}) \]

and

\[ \prod_{l=0}^{n} X_{j_{l+1}} \equiv Prob[\Phi(c^l) \land \ldots \land \Phi(c^{n+1})] \]  

(6)

\( \tau(\bar{u} \rightarrow \bar{x}) \) is the time necessary to go from \( \bar{u} \) to \( \bar{x} \) flying along the trajectory determined by the dynamics \( \bar{j} \), and \( \bar{u}^{l_i} \) is the process vector at the time of the initiation criterion setpoint for the deterministic transition \( \bar{j}_{l-1} \rightarrow \bar{j}_l \). Each non-zero term contributes to the right hand side of equation (5) exactly the same as each associated sequence in an equivalent static ET with headers \( \bar{j}_1, \ldots, \bar{j}_n \), and may be calculated from the cut sets of the front line systems using the classical PSA techniques [8]. However, the determination of the contributing paths is controlled by the step functions and \( \tau_n \) values, that depend on whether or not the initiation criteria have been exceeded. This analysis basically shows the correct way of understanding event trees and its relations to the combined \((j, \bar{x})\) evolution.

The two main conclusions are that the conventional sequence delineation rests upon the assumption of setpoint dynamic sequences, being then essentially conditioned by exceeding the settings, and on the limitation imposed by the relative time scales, i.e. relative short duration of the transients. Installations like NPPs have very complex automatic and manual actions, a fact that poses serious doubts about the possibility of decoupled \( \bar{j} \) and \( \bar{x} \) studies to delineate the setpoint sequences. In particular, it looks highly improbable the possibility of grouping initiators by the sequences they generate. Another conclusion of this analysis is the inadequacy of this sequence probability calculations in the case of transitions conditioned by alarm setpoints but with a stochastic time interval before the event actually takes place, operator actions being the most representative of those. In this short analysis the discussion of operators headers, as we call them, can not be included, but the reader is referred to [7] and [5].

3. Tools used in the ISA methodology

The ISA methodology, as inferred from the previous sections, proposes the dynamic generation of Deterministic Dynamic Event Trees (DDET) starting from an initiating event
that triggers the dynamics, and opening branches every time a setpoint or operator action
criterion is satisfied. Simultaneously, the cumulative probability of the sequence being
simulated must be calculated in order to obtain the desired results in terms of risk assess-
ment and to limit the size of the DDET. The development of software packages needed to
solve equation (5) follows the philosophy of coupling well known tools each one within
its own applicability domain. For data consistency, the basic tools we are coupling today
are the same as those being used standalone by the industry in Spain, tools for which
there are models for most plants, specially the severe accident code, hereafter referred to
as SAC. Traditional accident analysis models for the period prior to reactor trip, improved
to cover the post-trip period without core damage, that model in detail control systems
and automatic actions are used to develop the sequences. If conditions for its application
domain are satisfied during the sequence, the driver turns on classical PSA transient anal-
ysis tools currently used to delineate sequences and estimate the consequences during the
severe accident back-end. They have been coupled together (see 3.2 for the coupling ap-
proach) with fault tree style algorithms and PSA packages of the Saphire style to evaluate
equation (6). Research is being done as to precalculate the largest possible sections of
the large trees in a way that the effort that it takes to add new systems to the sequences is
optimized. Operator actions are simulated on a time step level interaction with the codes
with the help of suitable software packages.

The tools able to perform this kind of simulation must then be composed of

a. A scheduler that drives the simulation of the different sequences, storing the por-
tions of the tree that are common to several sequences in order to optimize the
simulation effort,

b. The simulation package, modelling the plant systems and the operator EOP actions,
with capability to evaluate the plant damage for each sequence,

c. The probability package, able to load the fault trees associated to each Event Tree
header, to perform their boolean product, and to calculate the probability of each
sequence.

The main features of these tools are described next. The scheme of the coupled tool can
be seen in figure 2.

3.1 Description of the scheduler

The event tree simulation scheduler is designed as a separate process that controls the
branch opening and coordinates the different processes that take part in the generation of
the DDET. The idea is to use the full capabilities of a distributed computational environ-
ment, allowing the maximum number of processors to be active, and avoiding to spawn a
number of tasks that would slow the computational effort. The distribution of tasks among
Figure 2: ISA plant simulation and operator modules coupling
the different processors is managed by the scheduler making use of communication software such as the pvm3 [9] suite.

To this end, it will manage the communications among the different processes that intervene in the event tree development, namely,

- the plant simulator,
- the probability calculator and
- an output processor in order to process the information concerning the results of each sequence.

The scheduler arranges for the opening of the branch whenever certain conditions are met, and stops the simulation of any particular branch that has reached an absorbing state \(^2\). The computation of new branches will not be started in case the starting point is already an absorbing state. To be able to decide on which branch is suitable for further development, the scheduler must know the probability of each branch, calculated in a separate process. Since the complexity of the probability calculation is different for each sequence, the traversing of the event tree will not be deterministic, but subject to the probability calculation time.

Each new branch is started in a separate process, spawning a new transient simulator process and initializing it to the transient conditions that were in effect when the branch occurred. This saves a substantial amount of computation time, since common parts of the sequences are not recomputed. The main drawback of this structure is that the output processing becomes a non trivial task, because a single sequence may be calculated by different processes. This is the reason why a separate process has been designed to build appropriately the relevant information for every sequence.

The applications of a tree structured computation extend beyond the scope of the DDETs. In fact, the branch opening and cutoff can obey any set of criteria not necessarily given by a probability calculation as, for instance, sensitivity studies or automatic initialisation for Accident Management Strategy following.

### 3.2 Description of the simulator tool

The engineering simulation tool is built in a modular form. Each module is a computational unit with input and output vectors, a control vector defining the different computational modes (steady state, etc.), a set of fixed data (not changing over the simulation

\(^2\)By this term we understand that the pair \((\pi, E)\) has reached either a point outside the damage criterion or a safe state.
run), a set of simulation variables and a flag indicating the state of the component. The module performs the computation in order to achieve the results at time \( t_n \), given that the input information is given at time \( t_n \) and \( t_{n-1} \) by means of any suitable numerical method. Modules can be of a fairly general nature, as for instance general ODE solvers, and of specific NPP phenomena and equipment, as centrifugal pumps or the pressuriser. The input structure of the simulation tool is formed by blocks, each of which is a particular instance of a module with specific fixed data, numbered so that the computation takes place in a specific order. The connection between modules is achieved by interconnecting the associated output and input. Both the connection and the ordering of the blocks define a topology upon which the global numerical method unfolds.

The modular nature of the simulator is additionally reflected by the fact that a module can be substituted by an external code that, in the extreme case, can perform the simulation of the whole plant model. To this end, the driver of the simulator is able to exclude some modules of the calculation of the plant model and insert dynamically new ones in replacement. This feature is used whenever the physical model that a particular module is able to solve falls out or is near the limits of its applicability range.

The nature of the connections is double: initial and/or boundary conditions can be provided to the connected codes. The most significant codes currently connected are,

**SAC** that performs the calculation of the plant model when the transient reaches a situation close to the exhaustion of the validity of its built-in models, because the transient approaches the severe accident conditions. It is then initialised with the appropriate transient conditions; some parts of the simulation (specially the operator actions, but also control systems) may still be performed by the original code and the appropriate signals be transferred as boundary conditions. A diagram of the connection with the SAC is shown in figure 3.

**RELAP5** [10] which can perform the simulation of a thermal hydraulic section of a simulation, receiving the boundary conditions from the tool.

**COPMA-II** [11] which follows the operating procedures and schedules the requested actions, sending signals to the plant model. Because of its design philosophy (as an operator training aid), the COPMA-II code is not able to perform an unattended simulation task, and is driven by an 'operator' that validates each action. Future developments of the code may permit unattended simulation.

Whenever convenient, provisions allow the different codes to run in a parallel environment when the problem so allows.
Figure 3: Plant Simulator – SAC connection
3.3 Description of the probability calculation module

The probability calculator module is the responsible of performing the logical product of the fault trees corresponding to each system that intervenes in the sequence either as a success or as a failure. Additionally, it will compute its probability as more fault trees are combined. This information is passed to the sequence scheduler. The fault trees that will be used for the probability calculations are those obtained in the development of the PSA results. This imposes a strong computational demand that is optimised by preprocessing the header fault trees as much as possible. Our current approach is to use the Binary Decision Diagrams (BDD) format, since it allows a compact representation of the logical functions associated with each fault tree and speeds up the computation of the products.

4. Applications

4.1 Procedure verification

One of the main applications of the ISA methodology is the verification of operating procedures. This has been tackled in two different on-going works, described next.

PWR SGTR In the first one [12], a sequence of events was triggered following a Steam Generator Tube Rupture in a PWR. The plant model includes a module, called Handbook of Operating Instructions (HOI) that models the operator behaviour in the course of the incident. During the accident evolution, failures of several components of the plant may occur. The evolution follows different paths depending on the configuration of the plant. Figure 4 shows the DDET obtained for the first 30 minutes of the SGTR scenario, after transitions due to interventions of four of the systems activated, either automatically or manually, namely, the Reactor Coolant Pump (RCP), the Steam Dump Control System (SCDS), working in both Average Temperature (ATCM) and Secondary Pressure (SPCM) control modes and the PZR spray valve (PS). The activation of these systems changes the plant status. In the tree shown, only some components of the status vector are considered: possible values of these status vector components are 0 for the nominal (unfailed) behaviour, 2 for RCP unavailability and when valves to the condenser (SDC), to the atmosphere (SDA) (excluding SG safety valves) and PS respectively fail to open upon demand. The mid-numbering appearing over the DDET branches indicates the demand time. Each path is numbered and each sequence is identified by the final status vector.

Figures 5 and 6 show typical variables for some selected sequences. The package is able to show the procedure steps evolution.
BWR SBO  The second application [13] concerns the sequence of events for the SBO initiator. In this case, the tools considered include a full description of the BWR EOPs, modelled with COPMA-II tool. The simulator is built in two pieces, one corresponding to the preaccident condition, with emphasis in the control and protection systems, while when the transient approaches the severe accident region, the simulation control is transferred to the severe accident code. COPMA-II is able to perform the EOP actions throughout the transient, irrespective of which code performs the plant simulation.

4.2 Accident Sequence Precursor Analyses

The nominal sequence will be made to correspond to the actual transient evolution of the plant in the event in accordance with the philosophy of NRC ASP program. Sequences, formed considering system additional failures, are then dynamically generated. By using this methodology, it will be possible to assess both the evolution of the plant during the event and the damage states that would be reached if the necessary equipment or operator actions fail, along with the conditional probability of each sequence. Additionally, the mapping of the event will be more accurate and many new type of lessons can be learned from it, for instance assessments of the settings of protection systems, as well as of the success criteria and sequence delineation of the PSA model.

Other applications  The general structure of the scheduler allows for any other applications that involve triggering transients at intermediate points. For instance, sensitivity studies may be done by this method of initialization under transient conditions when the result deviates from the expected trend, or when excessive error occurs referred to selected variables (Computerised Accident Management).
Figure 5: HPIS flowrate and Pressuriser level for sequences (1), (5) and (6)

Figure 6: Subcooling margin and steps evolution for sequences (1), (5) and (6)

References


Phenomenology and Course of Severe Accidents in PWR-Plants - Training by Teaching and Demonstration

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Abstract

A special one day training course on „Phenomenology and Course of Severe Accidents in PWR-Plants“ was developed at GRS initiated by the interest of German utilities. The work was done in the frame of projects sponsored by the German Ministries for Environment, Nature Conservation and Nuclear Safety (BMU) and for Education, Science, Research and Technology (BMBF).

In the paper the intention and the subject of this training course are discussed and selected parts of the training course are presented. Demonstrations are made within this training course with the GRS simulator system ATLAS to achieve a broader understanding of the phenomena discussed and the propagation of severe accidents on a plant specific basis. The GRS simulator system ATLAS is linked in this case to the integral code MELCOR and pre-calculated plant specific severe accident calculations are used for the demonstration together with special graphics showing plant specific details.

Several training courses have been held since the first one in November, 1996 especially to operators, shift personal and the management board of a German PWR. In the meantime the training course was updated and suggestions for improvements from the participants were included. In the future this training course will be made available for members of crisis teams, instructors of commercial training centres and researchers of different institutions too.

1 Introduction

Operators of German NPP’s are required to attend training courses on plant specific full scope simulators several times per year. These training courses covers normal plant conditions as well as training of accident situations together with existing preventive accident management measures and emergency operating procedures (EOP’s). The preventing of core melt sequences is trained intensively. For NPP operators up to now no training courses are required which covers e. g. the mitigation aspects of core melt sequences respectively the course of severe accidents.
Mainly due to the interest of German utilities GRS developed a special training course on „Phenomenology and Course of Severe Accidents in PWR-Plants“ in the frame of a project sponsored by the German Ministry for the Environment, Nature Conservation and Nuclear Safety (BMU). This is the first training course on this subject so far which has been held several times since the first one in November, 1996. Up to now the training course was primarily held to operators, shift personal and the management board of a German NPP with PWR. The feedback was very good and some impressions are discussed at the end of the paper.

It should be pointed out here, that there is no obligation by the NPP management board to their operators and shift personal to participate in the GRS training course. Nevertheless the number of participants is high and they are very interested on the subjects being presented.

2 Knowledge Basis

A wide spectrum of research work is ongoing at GRS sponsored by the German Federal Ministries BMU and BMBF and forming the basis of the development of this severe accident training course. In summary main contributions to the development of the different lectures of the training course are gained from:

- the research work performed at GRS in different accident management (AM) projects resulting in the development of AM measures, e. g.
  - prevention of RPV-failure under high pressure by secondary and primary bleed and feed measures,
  - prevention of over pressure containment failure by filtered containment venting procedure,
  - prevention and/or mitigation of consequences of possible hydrogen combustions in the containment by a passive autocatalytic recombiner (PAR) concept which is under development,
- the knowledge gained from several severe accident calculations performed at GRS within AM projects with the detailed codes ATHLET, RALOC and WECHSL and the integral code MELCOR using detailed PWR plant specific input decks,
- an extended literature investigation to define the level of state of the art on special subjects, where only limited research work is done at GRS.

Finally the development of such a training course is not possible without a powerful tool for the visualisation of different phenomena typical e. g. for severe accidents especially in relation to the detailed NPP behaviour. The ATLAS analysis simulator was developed by GRS with the aim to create a multi-purpose tool for analyses in the field of nuclear and industrial safety /BER 92/, /LUP92/. It is based on the GRS computer code
ATHLET modelling the dynamic processes in different plants /BUR 89/. In the past an extension of the system was made to make the system useful for different other codes /POI 96/, /SON 95/ and in many different areas of safety analysis too. The analysis simulator offers a simulation environment in which the presentation and evaluation of the numerous resulting data is supported by an interactive visual display system and which makes it possible to intervene directly in the simulation as it proceeds.

The power of the system will be demonstrated in this paper on examples showing the visualisation of a broad variety of MELCOR results for the primary and secondary circuit behaviour, the core melt process, the process of core concrete interaction and the containment behaviour of a German PWR plant used within the training course.

3 Intention of the Training Course

The main intentions of the training course are summarised in the following topics:

1. Development of a training course containing information on a state of the art basis and modern computer code based demonstrations about main phenomena of severe accidents related to German PWR.

2. Demonstration of a possible severe accident on an example (pre-calculated sequence with MELCOR) for the following main phenomena and phases:
   - initial plant conditions and early accident phase,
   - core heat-up and core melting,
   - core slumping into lower plenum and RPV failure,
   - molten core concrete interaction (MCCI),
   - fission product release, transport and behaviour in the containment,
   - hydrogen production (in- and ex-vessel) and release into containment,
   - hydrogen recombination/ignition in the containment,
   - long-term thermohydraulic containment behaviour.

3. Demonstration of the usefulness of different plant specific accident management measures respectively EOP's in different situations, e.g.:
   - the primary and secondary bleed and feed procedure,
   - different systems related to the hydrogen subject - the advantages and disadvantages of the use of autocatalytic recombiners or/and igniters,
   - the filtered containment venting system.

4. Sensibilisation of the audience for special severe accident problems, e.g.:
   - the instrumentation and information available during different stages of a severe accident to determine the plant status and to assess the accident progression,
the possible consequences of a very late injection of water into a hot or partly molten core as an additional accident management measure.

5. Development of a training course useful to held to a broad variety of people working in the nuclear field. Nevertheless the special interest is given to operators and engineers from German NPP's with PWR at the moment.

For completion it should be mentioned, that it is currently not the intention of this training course to focus on the detailed timing of different severe accident scenarios. Currently it is not possible for us to provide an "on-line" severe accident training course, running a severe accident code on a simulator tool and allowing interactions of the simulation by the participants. Reasons which are preventing such an on-line training are e. g. the level of development and validation and the robustness of the severe accident codes existing and the speed of such a calculation with a detailed plant specific input deck, which is several times less than real time. The necessity to held any kind of an "on-line" severe accident training will increase probably if special severe accident management guidelines are implemented in German plants.

Finally all these definitions above should clarify what we understand by training in the case of "Phenomenology and Course of Severe Accidents in PWR-Plants". It is really a seminar to increase the knowledge about severe accidents and to make sensitive to special problems and aspects of them.

4 Overview of the Training Course

The training course is subdivided into three parts. These are an introduction into severe accidents and the used simulator system, phenomena related to in-vessel phase and phenomena related to ex-vessel phase. In each part of the training course short presentations of basic knowledge alternate with supporting demonstrations by the ATLAS simulator. Within the frame of this presentation it is not possible to discuss all the details of the one day training course. Therefore we would like to focus only on main subjects.

4.1 Introductory Chapter

In the introduction some information about the general safety concept are given first. Even if the basis of this concept is well known it seems to be necessary to replay some of the main topics and to generate a basis for further discussions. Further topics presented are e. g. the use of different plant specific accident management measures to prevent core damage or to mitigate the consequences and the probability of different core damage states. In the second part of the introduction the used simulator system ATLAS and the used code system MELCOR is presented. Special attention is given to the discussion of the advantages and disadvantages of the integral code MELCOR together with the presentation of the plant specific nodalization schemes used.

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4.2 Phenomena related to In-Vessel Phase

4.2.1 Initial Conditions and Early Accident Phase

The boundary conditions and assumptions of the scenario „Total loss of steam generator feed water supply and loss of heat sink“ used as example for the training course are discussed first. The intention was to use a scenario with a relevant probability and a RPV failure at low pressure, where the most of the relevant phenomena of severe accidents can be demonstrated and where the use of existing accident management measures (secondary and primary bleed and feed, filtered containment venting) can be demonstrated as close as possible following the instructions of the existing EOP’s. For all the sequences discussed the failure of high and low pressure injection systems was assumed from the beginning. These assumptions leads to the earliest time of core melting but causes also a lot of discussions from the audience.

The attention during the presentation of the early accident phase was given to actions e. g. of functions of the volume control system, the time dependent behaviour of main parameters and initial criteria for the used bleed and feed measures, well known by the NPP operators. So they can familiarise with the different kind of graphical presentation of plant specific parameters by the ATLAS simulator too.

The examination of the needed information and the available instrumentation to determine the plant status showed, that the normal plant instrumentation is challenged by the containment conditions (temperature and humidity) soon after the depressurization of the primary circuit. Those instruments are not qualified for the expected containment conditions. Therefore only a limited number of information are available mainly from the safety-related and the wide-range instrumentation installed.

Besides some available plant specific graphics, the main graphic used for the demonstration is the one named „PWR Reactor Circuit“. Figure 1 shows one example at a time of normal plant operation. A large number of dynamic effects are included in this graphic as well as in all others. In this case the water temperature dependent on a pre-defined colour scale is shown in all parts of the reactor circuit with exception of the core. But this is only one example of the large number of physical parameters which can be selected immediately coming from the MELCOR simulation.

To allow the demonstration of the core melting process together with the reactor circuit behaviour the detailed fuel element (\(\text{UO}_2\)) temperature is shown in each cell of the MELCOR core nodalization scheme dependent on a separate colour scale too.
Figure 1. ATLAS graphic „PWR Reactor Circuit“ - Behaviour of Reactor Circuit during Normal Plant Conditions

4.2.2 Phase of Core Heat-Up and Core Melting

The discussion of phenomena of core melting is one of the main interesting parts of the training course. Because of the wide spectrum of existing information and experimental knowledge about the core melting process it becomes important to focus only on topics supported by examples from the TMI accident and from typical experiments of the CORA facility of FZK and from the PHEBUS facility of IPSN.

In relation to the course of the core heat up phase the early destruction of control rods by different phenomena including eutectic reactions is discussed first. Principles of the candling process of the molten material and the solidification and possible formation of local blockages in lower, colder elevations inside the core are presented than. After that the principles of the fuel element failure mechanisms are discussed and especially the oxidation process of the fuel element tubes and the H₂ production is shown. At the end we discuss the formation of a debris pool in the lower core region supported by information from the TMI examination and from CORA and PHEBUS experiments.
It is well known, that the computer simulation of the core melting process by an integral code can be done only with some limitations. Therefore the audience is informed in each part of the training course about the simplifications of the code system used related to the special accident phase. Nevertheless based on the feedback from the first training courses we think, that the developed and used graphics gives equivalent support to the complicated subject of the core melting and relocation process. The main graphic used for the demonstration is the one named „PWR Reactor“ (Figure 2).

![Figure 2. ATLAS graphic „PWR Reactor“ - Begin of Core Melting Process](image)

The example in Figure 2 shows the situation at a time just before the beginning of core (fuel element) melting and relocation. The core heat-up process as well as the heat up and candling of control rods just before fuel element melting can be demonstrated quite well. The relocation of the fuel element material and the formation of a debris pool can not be demonstrated in the same way mainly due to code (MELCOR) limitations.

Finally the available instrumentation to determine the plant status is discussed. It is very likely, that during the core heat-up and core melting phase all instruments located in the primary circuit may be experienced by temperature conditions beyond their qualification limit. Therefore the determination of the exact plant situation seems to be impossible.
Further investigations are necessary to assess the influence of those high temperatures inside the primary circuit to measurement techniques installed there.

### 4.2.3 Phase of Core Slumping into Lower Plenum and RPV Failure

Based on knowledge gained e.g. from the TMI accident and from experimental investigations the events and phenomena during core slumping, molten pool formation inside the lower plenum and the RPV failure mechanisms are described.

As in the chapter before only some topics can be explained, e.g. the mechanisms of material relocation due to local or global failure of a molten pool above the core support plate, the quenching mechanisms of the molten material in the residual water inside the lower plenum, the potential of high energetic in-vessel steam explosions and the different RPV failure mechanisms depending e.g. on the design of the vessel and the system pressure. In addition an overview on expected loads to the reactor cavity and the supporting concrete structures dependent on the RPV failure mode respectively the system pressure is given. Finally the ongoing experimental and theoretical research work on the subject of cooling of a molten pool inside the RPV bottom head by external flooding of the reactor cavity is discussed.

During the first hours of such an accident a large number of instrumentation is expected to be failed. No special instrumentation is available to determine the RPV failure. Some information can be taken from the pressure measurement of the reactor circuit and the containment. There should exist a difference between both pressures until the RPV fails. Nevertheless it is not well known, if the pressure measurement of the reactor circuit will fail or not before.

Mainly due to code limitations (MELCOR) the possibilities to visualise the physical phenomena (material relocation, steam explosions, molten pool behaviour in the lower plenum) are rather limited. Therefore we focus on the pressure and the oxidation behaviour and the \( \text{H}_2 \) production during melt relocation and quenching. Thus the ATLAS graphic „PWR reactor circuit“ together with history diagrams of different parameters are used.

### 4.3 Phenomena related to Ex-Vessel Phase

#### 4.3.1 Phase of Molten Core Concrete Interaction (MCCI)

The reactor cavity of German PWR containments consist of an inner cylindrical biological shielding surrounding the RPV and an outer thick cylindrical support structure, made both of reinforced concrete. It is typical for German PWR containments that the reactor cavity is dry. Thus a dry MCCI reaction is expected during the first hours after the RPV
failure. Due to radial erosion of the concrete a sump water ingestion into the reactor cavity is expected to occur some hours after RPV failure followed by a long-term evaporation of the incoming water. Very special plant specific details of the cavity design (inspection openings, air ventilation pipes) have not been included in the research so far, leading to discussions with the audience during the training courses.

Figure 3. ATLAS graphic „PWR Cavity“ - Molten Core Concrete Interaction

A simplified graphics shown in Figure 3 is prepared showing a part of the reactor cavity where the erosion front behaviour can be demonstrated. Figure 3 shows a situation some hours after RPV failure but in any case before sump water ingestion. The temperature of the melt and the atmospheric temperature inside the rooms as well as the sump water level is shown. The erosion front behaviour as well as the thickness of the melt can be expected from the axis included in the graphic.

4.3.2 Fission Product (FP) Release and Transport in the Containment

Within the limited time frame of such a training course only a well selected overview on important phenomena related to the FP behaviour can be given. The presentation focuses first on the initial FP core inventory. The different phases of the FP release and transport during core melting are discussed together with some influencing factors. The time dependent concentrations of aerosols and noble gases released can be shown by
different graphics of the ATLAS system. The example in Figure 4 shows the release of air born aerosols during core melting and the transport through the primary circuit.

![Figure 4. ATLAS graphic „PWR Reactor Circuit“ - Aerosol Release and Transport inside the Primary Circuit during Core Melting](image)

Simultaneously the time dependent behaviour of aerosols and noble gases in the containment can be shown by the ATLAS simulator. One example is given in Figure 5, showing the transport of air born aerosols inside the containment released from the primary circuit during core melting.

Typical phenomena related to the general FP behaviour are discussed too, as e.g. sedimentation, absorption, condensation and deposition of FP. Also some information are given to the importance of Iodine and its behaviour in comparison to other aerosols. So the theoretical presentation of the aerosol behaviour in the containment is supported by this graphic. The same graphic is used to demonstrate the FP release from the reactor cavity during the early phase of MCCCI in the reactor cavity.

At the end of this part of the course some results of plant specific calculations of FP release rates into the environment are presented. The importance and the influence of FP retention in the primary circuit, in the containment and in filters of the containment venting system on the release rates into the environment becomes obvious and further-
more the importance of a late initiation of filtered venting too. It can be demonstrated that the noble gases are the main contributor to the source term into the environment. These elements can not be retained by the filters of the venting system. So about 75% of the noble gas inventory is released into the environment.

Figure 5. ATLAS graphic „PWR Containment“ - Aerosol Transport inside the Containment released from Primary Circuit during Core Melting

4.3.3 Hydrogen Production, Release and Behaviour

As for the selection of topics on FP behaviour it is very important to concentrate only on main phenomena of the large area of hydrogen behaviour. Several specialists meetings and workshops have been held world wide on this general topic and a large number of experimental research is ongoing. In Germany these general topic is very important because extended research work is ongoing on the development of a PAR concept for German dry PWR containment's. Therefore the actual situation in Germany and the extended research work at GRS is characterised and preliminary results of calculations with the detailed containment code RALOC for a German PWR containment are presented. The containment behaviour during a selected severe accident is shown with and without an assumed distribution of a large number of PAR's. It becomes obvious, that
such an PAR system is very useful to remove large amounts of hydrogen within an acceptable time frame. The prevention of combustible conditions by this system inside the most of the containment compartments can be shown. Local combustions are expected only during special situations. They will not influence the containment integrity.

In addition special attention is given during the presentation on the main phenomena of hydrogen generation during core melting and MCC1. Basic knowledge about deflagration, DDT and detonation phenomena and the related challenges to the containment integrity are presented too.

The demonstrations by the ATLAS system supporting the information of this chapter are made with the graphics showing the „PWR containment“ (see example in Figure 5) and using the hydrogen mole fraction as selected parameter. An additional graphic is used to determine the flammability of the gas mixture in for selected rooms of the containment (Figure 6).

![Figure 6. ATLAS graphic „Flammability Limits“ - Determination of the Situation in 4 different Containment Rooms and History of Total Hydrogen Generation](image)

Due to limitations of the MELCOR code no demonstration is possible on the influences of a PAR system on the containment behaviour. Only if there is enough time remaining
during the training course, a demonstration is made showing some influences of local hydrogen combustions on the containment loads. A pre-pared MELCOR calculation exists in this case for the same accident scenario including the calculation of hydrogen combustions initiated by assumed igniters.

4.3.4 Long-Term Containment Behaviour

The final part of the training course deals with the containment behaviour (temperature, pressure) and the use of the filtered containment venting measure. Special attention is given to the phenomena and events influencing the containment pressure during all phases of the accident. The example in Figure 7 shows the pressure history (trace back from actual time by 6 hr) in the primary and secondary circuit as well as in the containment and the annulus.

![Graph showing pressure behaviour](image)

**Figure 7. ATLAS history „Pressure Behaviour“ at different Locations**

In the history diagrams in Figure 7 the influence of the secondary bleed and feed measure on the primary and secondary circuit pressure (measure initiated at -5 hr) becomes obvious. The influence of the primary bleed measure on e. g. the containment pressure can be seen too (measure initiated at -2 hr).

Information about the energy respectively steam release into the containment at different locations (pressurizer relief tank, cavity) during all phases of the accident are also included to demonstrate e. g. the temperature behaviour inside the containment. The example in Figure 8 shows the steam release into the containment after the depressurization of the primary circuit. It is interesting for the audience to notice that during some
time intervals large differences in the concentration of physical parameters (e.g. steam, non-condensable gases, aerosols, noble gases) inside the containment are expected.

Figure 8. ATLAS graphic „PWR Containment“ - Steam Release from Pressurizer Relief Tank into the Containment after Primary Circuit Depressurization

Last but not least this final chapter summarises the main contributions to the containment loads. The importance of the filtered containment venting system to prevent an over-pressure containment failure is underlined too. Finally the critical situation regarding the available instrumentation and the needed information during severe accidents is repeated ones more.

5 Feedback and Summary

In the previous chapters of the paper the intention and the main subjects of the training course on „Phenomenology and Course of Severe Accidents in PWR-Plants“ developed by GRS are discussed. It becomes obvious that we may interpret differently the word „training“ depending on the subject being presented in the training courses. For training courses on severe accidents it is currently not possible for us to provide them „on-line“. running a severe accident code on a simulator tool and allowing interactions of the running simulation by the participants. Main reasons which prevent such an on-line training
are the level of development and validation of the existing (integral) codes, their robustness and the time which would be required for such a training. The need for any kind of on-line severe accident training may be different in other countries, dependent e. g. from the implementation of severe accident management guidelines in the plants.

The lessons we learned from our training courses being held are described in the following without any hierarchical order.

1. The participants from various departments of the NPP were interested in the general subject of „severe accidents“ but their special interest varies depending on the scope of duties.

2. Nearly all of the participants underlined that the training course becomes so interesting due to the mixture of short compact presentations and informative demonstrations by the ATLAS simulator system used.

3. The time frame of one day is very short to present all main topics of the wide range of severe accidents including an extensive demonstration by the ATLAS simulator. Therefore the subjects of the different chapters of the training course should be well balanced.

4. We would prefer to present the materials in the future on a 1 ½ or 2 day training course. We thus would split the course into the areas of in-vessel phenomena (first day) and ex-vessel phenomena (second day).

5. The training course described here is presented by two persons working in the area of severe accident research for several years. A detailed knowledge on most of the subjects presented and on plant specifics by the lecturers is necessary to get confidence from the audience and to answer the different questions. The confidence in the results presented rises also due to the discussion of disadvantages and limitations of e. g. the used simulation tools.

6. Examples of topics from the discussions during the courses are the following:

   - Why do you assume the failure of all safety injection systems? (done for the scenarios used as examples)
   - What are reasons for large differences between the probabilities of similar events in different plants shown in the introduction of the training course?
   - Would you recommend to inject water into RPV at any time during core melting before RPV-failure?
   - In which way does the plant specific cavity design may influence the course of MCCI and the containment behaviour?
   - What do we notice in the control room from the different processes in the containment during severe accidents?
For the near future it is planned to present this training course to instructors from a commercial NPP training centre and to researchers working for PSA level 2. At the beginning of the next year additional training courses are under discussion at further German NPP's. An application of the training course to the behaviour of German BWR under severe accident conditions is planned too.

**Literature**


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An Interactive Graphic Simulator for Garoña Nuclear Power Plant: Development and Applications

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Abstract

As a result of a R&D project jointly undertaken by Nuclenor S.A. and the University of Cantabria, since the beginning of 1995 Santa María de Garoña NPP holds its own plant transient and accident simulator for the training of its operators on Emergency Procedures Guides (EPG’s). This paper describes the process followed in its development and the derived projects.

This interactive graphic simulator is based on the MAAP computer code with an interface developed with DataViews® software for X-Window terminals running under UNIX on a Hewlett Packard series 9000/712 workstation.

The thermo-hydraulic code MAAP, with modifications to include all the controls and systems of the plant, was chosen due to its capability to simulate primary system, containment and reactor building behavior during transients and accidents of any kind, including those beyond the design basis and also severe accidents. Besides, this code is currently being used for the PSA of the plant.

The current configuration consists of several monitors that displays the systems and components situation and the safety parameters through the SPDS (Safety Parameter Display System) displays. With this design the simulator has fulfilled its initial main purpose of training operators on abnormal and emergency procedure guides, revealing as a very efficient and cost effective tool to accomplish this task.

Furthermore it has been improved to visualize the real time plant data through the simulator displays. It is also possible to distribute the simulated data to any computer connected to the local area network (LAN). This feature has been applied successfully in
the last emergency drills. Other benefits are derived from the fact that the same code is used for training and for the PSA.

As an example of its flexibility, the simulator is now being used to validate the new SPDS displays designed to improve the man-machine interface and then it will serve to train the operators on them before their installation at the Control Room.

Finally, reference is made to a set of different future projects, including validation of the new guidelines related to severe accident management, linkage to a dose calculation code, and incorporation of an expert system to track the EPG's.

1. Introduction

This paper describes the results of a series of research and development projects in the area of simulation and plant data management that have been carried out for Santa María de Garoña NPP during the last years. It is important to observe that Garoña plant, a General Electric Boiling Water Reactor with a nominal thermal power of 1381 Mw, has been connected to the grid since 1971, being the second oldest Spanish nuclear power plant and, therefore, having accumulated a large experience in all these years of operation.

Most of these projects, as the Interactive Graphic Simulator, have been developed jointly by Nuclenor S.A., the owner of Garoña NPP, and the Department of Applied Mathematics and Computer Science of the University of Cantabria. As we will see later, the benefits derived from them have outdone by far the design objectives, revealing useful for many other applications. In this way, Garoña NPP currently holds the suitable tool for accomplish most of the training requirements that Severe Accident Management imposes.

The first section of the paper includes a brief explanation of the simulation background that led to the realization of these projects. Then, the Interactive Graphic Simulator is described in detail: simulating code, graphical interface, configuration and current uses. The next section refers to the Real Time Plant Information System, its relationship with the simulator and the advantages of having integrated both tools. Finally, a set of future projects and feasible applications, some of which already started, is shown.

2. Background

From 1984 up to these days, the Spanish nuclear power plants have been performing Probabilistic Safety Analysis (PSA) of a continuously increasing scope. The first of these studies was the Santa María de Garoña Level I PSA as required for the
Spanish Nuclear Safety Council (CSN). This study, finished in 1985, was performed with limited information on plant transient behavior and without any simulation tool.

Since 1987 we have been working on plant simulation models aiming to study, analyze and develop information science techniques for the control and simulation of the systems which make up a nuclear power plant. Our interest was focused in the fields of Neutronics and Thermalhydraulics with specific application to Garoña plant.

Taken into account that codes capable of simulating the running of a nuclear power plant such as Garoña were already available, the first stage was to develop a set of applications that, by incorporating graphic capacities, would enable a faster interpretation of the results of the simulations obtained by these codes, mainly LTAS, TRAC and MAAP. These applications were designed in such a way that real plant events could also be analysed and even compared to the corresponding simulations.

With the experience acquired from these works, it was decided to develop a training simulator which, while maintaining, and also improving, the previously mentioned graphics features, would allow an interactive link to the adopted simulation code.

Lacking of a full scope simulator, Garoña's operators are trained at the Monticello NPP full scope simulator, so the main requirement to the new graphic simulator was to be able to cover at least the operator training in Emergency Procedures Guidelines (EPG's), saving an important amount of training hours at Monticello. For achieving this goal it was necessary to develop an interface through the SPDS (Safety Parameter Display System) displays with the interesting consequences that are described later.

3. The Interactive Graphic Simulator TSG

TSG (Training Simulator for Garoña Plant) is the Interactive Graphic Simulator that resulted from that project. It was released at the beginning of 1995 after a thorough validation against actual plant events. The CSN acknowledged it as a valid tool for training and, since then, Garoña's operators spent a fixed amount of hours training on it every year.

The simulator is capable of simulating the plant behavior, in real time or much faster, during abnormal and emergency events, including those beyond the design basis and also severe accidents. In addition to some specific displays, the results of the simulation running are shown through the same SPDS displays that are used at the Control Room.
3.1 The simulator core: the thermalhydraulic code

In order to choose the most appropriate code to build this interactive simulator, the TRAC and MAAP codes were studied in depth: structure, modelisation capacities, accuracy, speed of calculation, suitability for interactive use and so on. Finally, MAAP 3.0B was selected as the inner code of the simulator. Being an integral code, capable of simulating much faster than real time the thermalhydraulics of the primary system, containment and auxiliary building with all the associated systems in normal operation or during any kind of accident or malfunction it seemed to match perfectly with the simulator requirements. Its degree of accuracy was high enough for its main purpose of training as the complete test of transients compared to actual plant behavior proved later. Moreover, regarding its application to severe accident management the code has been specially designed to carry on the sequences after core melting and calculate the source term.

Some others reasons favored this decision. In 1995 the CSN required a revision of the Garoña Level I PSA that had been performed ten years before and MAAP 3.0B was selected to support the accident sequence analysis part of the study. The advantages of using the same code for training and PSA are described later.

The modular structure of MAAP constituted an additional help for the development of the simulator. Besides the easy way that provided to link the code with the user interface, it allows to improve some physical models or add new routines with little effort. Therefore, the original MAAP has been adapted to the training requirements. Some modifications were needed to permit an interactive use and a series of other changes have been performed until now, including:

- Feedwater and pressure control modules based on the TRAC model.
- Introduction of a simplified kinetics model by means of correlations with the water temperature and the recirculation flow. It is also possible to modify the total thermal power while the sequence is running to simulate manual control rod insertion.
- Splitting of some systems that the original MAAP treats as a single loop, like the Core Spray, Control Rod Drive System, Stand-by Liquid Control and Shutdown Cooling System.
- Possibility of changing gradually a tube rupture area for a period of time during the simulation.
- Introduction of new user events to represent either operator actions or added malfunctions.
Provided that they are the main users of the application, many of these modifications have been developed taken into account the instructors and operators suggestions after each training period. In this way the simulator is evolving continuously. It is supposed that future training on Severe Accident Management will imply new changes but the acquired experience assures us that they will be easily implemented.

### 3.2 Simulator architecture: the interface

The simulator interface was designed according to the following criteria: it should result a friendly application, with direct communication with the inner code, easy of improving and portable to different machines. The Dataviews software, a real time graphic application, was chosen due to its orientation towards instrumentation engineering and process control. In addition, this software uses the graphics standard X-Windows (X11) running under UNIX operating system and can be executed even on PC platforms.

Therefore, the simulator consists of several graphical screens that make up a user environment. The plant systems and components are represented by objects whose dynamic properties are associated to the events and variables which determine their performance. A program written in C language and linked with the needed DV-Tools routines from the Dataviews package manages the human-machine dialogue, while the FORTRAN MAAP routines perform the thermalhydraulics calculations. At the beginning of each session a routine creates the connections among the objects properties and the matching MAAP events and variables. The communication between both programs is maintained as long as the simulation runs using a shared memory area that includes the needed MAAP COMMON blocks.

Several users can interact simultaneously with the simulator. It is possible to create malfunctions or make the plant systems respond to such anomalies clicking with the mouse on the buttons or sliders of the corresponding screens. The remaining screens are designed as a help to the operator displaying information about the sequence evolution and the systems situation. The users have the possibility to navigate among these last screens which include the SPDS displays.

In order to facilitate the training of the operators two auxiliary tools have been developed for the managing of the input and output data. Instead of writing a MAAP input file for each scenario, the instructor has a graphical application available for the automatic generation of the MAAP input code in such a way that he can set the initiating events and operation conditions at the beginning of the sequence just using the mouse. In the same way other application allows to plot any output variable to analyse.
the results of the exercise at the end of each simulation. Both tools have also been developed with Dataviews® software.

3.3 Current configuration

At the moment, the simulator layout at the plant training building consists of six monitors (X-11 terminals) as shown in figure 1.

![Figure 1. Current disposition of the simulator at Gorloša NPP training building.](image)

One of them constitutes the instructor’s console from where he can define the scenario to run, cause new malfunctions in the course of the simulation and control the evolution of the sequence (real time or faster, backtrack or restart). The operator has his own console, which all the plant safety systems represented, to interact in real time with the simulation and try to manage the plant behavior. He can start or stop systems, adjust flows, open or close valves and so on.

The others four monitors serve to visualise the plant situation and help the operator in his task. One of them shows a group of alarms, resembling the actual alarm panels at the Control Room although only the most significant ones are represented. A sound signal warns the operator each time a new emergency signal is detected. Other monitor displays a set of messages describing in detail the actions taken by the operator and the response of the plant itself. In the remaining two monitors several displays can be selected to observe the plant evolution and the systems state.

Among the available displays for these last monitors there are the SPDS ones. The reason is obvious as stated above in this paper. The Interactive Graphic Simulator must serve to train operators on Emergency Procedures Guides and so, the same information should be used during training and during operation. However, the
relationship between the simulator and the SPDS is stronger as will be described in the following section.

Currently, the simulator is running on a Hewlett Packard series 9000/712 workstation with 100 MHz processor. For training purposes the system runs in real time but for other applications it may run even twenty times faster than real time, depending on the sequence. The hardware that constitutes the simulator layout also includes one laser printer for printing graphics of the evolution of the desired variables during or at the end of each session. An appealing feature of the simulator is that the machines in which it is installed do not need no be particularly powerful; low priced workstations and/or high level personal computers are enough.

Figure 2 represents the system hardware as well as some typical displays of each monitor.

![Figure 2: Interactive Graphic Simulator layout.](image)

**3.4 Current uses and derived benefits**

The main purpose for which the simulator was designed, training of operators on EPG's, has been fulfilled. Periodically Garoña's operators receive courses on EPG's following the SPDS displays at the Interactive Graphic Simulator, reducing the amount of training hours at the Monticello full scope simulator that otherwise would have been required. So, the simulator has revealed itself as a very efficient and cost effective tool.
Furthermore, several different applications, some of them unexpected, have been found since it was released. For instance, it has become a very useful aid to the engineering staff responsible for the PSA simulations. Apart from the advantages of being able to analyse any PSA sequence, or some period of it, in real time while it is running on the Interactive Graphic Simulator, it has been used to verify all the new models or improvements added to the MAAP model for the PSA study. It is known that some of the lots of physical phenomena handled by MAAP are not modelled with the required detail. Thus, many times they raise some kind of problem that can only be solved by modifying the original code. Following this procedure it is granted that all these changes will be readily included in the training simulator model.

Moreover, the use of a common tool for training and safety analysis favors the collaboration between the personnel responsible for these two areas. In this way, some human reliability aspects of the PSA, especially those relating to operator response, have been better analysed with the help of the simulator, discussing the evolution of the sequence and the operator actions with the operators themselves or their instructors.

The Severe Accident Management programme that is being currently developed for the Spanish nuclear power plants considers the training on severe accident scenarios. At the same time, Garoña Level II PSA study will be performed and so, the same profitable collaboration will be exploited.

Finally, there are supplementary benefits derived from the simulator output through the SPDS displays but they will be exposed in the next section because of their relationship to the Plant Information System.

4. Real Time Information System

The varied set of parameters which affect the plant operation (process data, control rod monitoring, water supply or meteorological data for example), are acquired by several systems with different purposes. These systems have been installed gradually during the operational years of Garoña plant and now constitute an heterogeneous data access system that uses diverse operating systems and with its output consoles located exclusively at the Control Room or at the Process Computer Room.

From 1988 Nuclenor has been working on the computerized management of plant data, mainly focused on plant operation personnel. In this way, the SPDS was developed to substitute the software included in the process computer for data display, initially as a closed proprietary system that only reached the Control Room. The Dataviews® package was also chosen due to its characteristics as process surveillance software.
After the development of the Interactive Graphic Simulator and with the gained experience, it was decided to implement a new application to distribute in real time the SPDS displays through the company's local network, facilitating the access of all the personnel to plant data. It was released in the middle of 1995.

### 4.1 Description of the system and evolution

This new application, the SPDS-D (SPDS-Distributed), benefits from the open protocol that represents the standard X11. So, any computer connected to the Nuclenor network, with the unique requirement of being a X-Window station or a PC with X-emulation software for Windows, can access to the real plant data contained within the SPDS displays and navigate among them in real time. Besides, as the software runs in a number of different computer systems is not likely to be a limitation when considering alternatives in computer hardware. Additional features include plotting of trends of selected parameters, printing graphics of their historical values or the access to the additional displays which were developed for the TSG, with their corresponding data from the actual plant situation.

The next step in the evolution of the system was logical. As long as the same displays were used as output of the simulations executed on the Interactive Graphic Simulator it was easy to integrate both tools in such a way that now, from any computer connected to the network, it is possible to feed the SPDS either with real data or simulated data. It is also feasible to visualise previously recorded SPDS data.

The integrated system layout is shown in figure 3. As can be observed, the company holds a local area network (LAN) with Ethernet links, optical fiber and a microwave link between the plant and the corporate offices which are 140 km away. The plant is also linked to the Emergency Room (SALEM) at the CSN in Madrid.

Nowadays, Nuclenor is in the process to configure the Real Time Plant Information System in a more simplified manner. Plant data captured by all the acquisition systems will be send to a Data Concentrator-Distributor (CDD). This machine, an UNIX computer, will process the information as required and then act as a server sending the processed data to the client applications. The simulated data from the TSG, which is being configured as a client/server application too, will also be sent to the CDD allowing that the SPDS clients may switch between real and simulated data when desired.

At the same time, the SPDS pages are going to be replaced with a more modern design, similar to the Monticello’s SPDS. As said above, Garoña’s operators must use those pages during its training at the full scope simulator and, besides, they incorporate
a series of interface improvements resulting from the latest standards and studies on human-machine interface. This new SPDS will constitute the first client application of the new information system.

Figure 3. Integrated system layout.

4.2 Related TSG applications

With its present configuration, the integration of the Interactive Graphic Simulator with the SDPS-D application has produced substantial benefits. The simulator served to validate the first SPDS pages with the needed dynamic approach due to the slight variability in the visualised process during normal operation. Moreover, since their installation at the Control Room, the generation and maintenance of operator interface software has been common for the plant data system and the simulator.

However, the most attractive application of the distribution of simulated data through the Nuclenor network has been experienced in the last emergency drills. These exercises are carried out periodically at the plant, aiming the training of all the personnel on emergency response. Since 1995, the accident scenario prepared for the drill runs on the TSG. In this way, the simulated data is displayed simultaneously on the Plant Technical Support Center and on the Emergency Response Center at the Corporate Office. At the same time, the required data is sent to the CSN’s SALEM in real time directly from the simulator. This capability has obviously introduced more realism in the drill contributing to a better training of the personnel.
Regarding the substitution of the current SPDS, the Interactive Graphic Simulator has been recently applied to the dynamic validation of the new pages, as done with the first SPDS. A series of sequences have been executed on a new version of the TSG, configured to send data to the CDD, to verify the proper working of the system. To remark the simulator significance at Garoña NPP, it is interesting to note that this validation has been required as a compulsory previous step before the installation of the new SPDS at the Control Room. Meanwhile, the operators will get used to the new displays training on this version of the TSG that will become the definitive one.

5. Work in progress

As shown in the previous sections, the simulator has been evolving continuously since it was released, taking into account the users suggestions and the company requirements. Nowadays, a set of potential future applications are being studied and some of them have already started its development. In this section we explain briefly the most relevant ones.

5.1 Severe Accident Management

As said above, in a short time the training of operators on severe accident scenarios will be required to all the Spanish nuclear power plants. It will be easy to adapt our simulator for accomplishing this task, introducing some minor changes on the interface and the modifications of the MAAP code implemented during the Garoña Level II PSA that is being started these days. The simulator could serve to validate the Severe Accident Management Guidelines (SAMG's) that will be developed.

We are also considering the possibility of upgrading the simulating code from MAAP 3.0B to MAAP 4, with its enhanced capabilities regarding severe accident. Nevertheless, having completed the Level I PSA with MAAP 3.0B it seems more reasonable, at least while the Level II is performed, to maintain the present version that benefits from all the changes already implemented and the achieved experience of the engineering staff.

A more ambitious project deals with the potentiality of linking the MAAP code to a dose calculation code (e.g. ORIGEN). This tool could be used for the future Level III PSA and, if the simulator were adapted to use it as its inner code, it will become an integrated application that would cover all the stages of any kind of conceivable accident.
5.2 Additional interactive displays

Nowadays we are modifying the simulator control model in order to obtain several additional interactive displays for the operator, as a replica of the real panels present at the Control Room. Currently, the operator acts on the simulated systems by means of sliders and buttons using a simplified display. The new detailed displays will contain all the switches, buttons and controls of the real panels, adding more realism to the operator actions. The final purpose of this project is to replicate every panel present at the Control Room in such a manner that the training at the TSG will resemble the training at the full scope simulator.

5.3 Expert systems

The simulator could incorporate an expert system for tracking the EPG’s, assisting the operators with its indications to manage emergency situations. This system could be configured as a client application of the new plant information system. Moreover, the expert system does not need to be restricted to give advices; it could be implemented in such a way that the transient evolution run on the simulator would reflect the actions that the operator should apply according to the EPG’s, working as an ‘automatic operator’. This last approach could be used as a very valuable tool for developing and testing the future SAMG’s.

A plant data analysis module would allow the generation of a MAAP input file reproducing the actual plant situation. In this way, and given that it can run much faster than real time, the simulator could be initialized at any point in time into a plant transient serving as an operator aid. The module for tracking the EPG’s would reinforce this support.

6. Conclusions

The development of the projects described in this paper has supposed a very profitable experience, not only to the personnel involved in the developing process, but also to all their users during the time they have been working. We must remark the needed effort to adjust the simulating code to our plant and to distribute the results across the network in a graphical manner and real time and, consequently, the great experience acquired with this tasks.

As a result of all this work, now Nuclenor is capable of carrying out an important aspect of the training of Garoña’s operators at the plant itself and holds a modern plant information system with a constantly growing range of applications. Future
requirements in any of these areas could be undertaken with sufficient assurances of success.

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Coupling of 3D Neutronics Models with
the System Code ATHLET

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Abstract

The system code ATHLET for plant transient and accident analysis has been coupled with 3D neutronics models, like QUABOX/CUBBOX, for the realistic evaluation of some specific safety problems under discussion. The considerations for the coupling approach and its realization are discussed. The specific features of the coupled code system established are explained and experience from first applications is presented.

1. Introduction

Analytical simulation methods are an important tool for the safety analysis of NPPs. Efficient computer codes have been developed for each physics area. The details of models applied differ according to the requirements in accuracy and speed needed for single effect or integral effect analysis. In most cases a family of models is available to meet the
specific requirements. The different weighting of importance for accuracy and speed of simulation, respectively, determines the existing differences of models applied in safety analysis and simulator applications. A very general experience of simulation is the permanently growing computer capacity which is still ongoing. This continuously diminishes the need for simplified models, because more accurate models based on complete sets of physics equations can be solved sufficiently fast.

On this background, the need of analysis for specific safety problems initiated in the field of 3D reactor core models and in the field of plant transient analysis the development of coupled codes. Such codes combine transient 3D neutronics models for reactor core analysis with large best-estimate codes for plant transient analysis. The development of these codes has reached a very high interest leading to special sessions related to this topic within several international meetings /1, 2/.

2. Current Safety Problems requiring Application of Coupled Codes

In the past, safety analysis was based on the application of 3D reactor core models to analyze local power density distributions and reactivity conditions of the reactor core using specified boundary conditions, and on the application of system codes to analyze plant transient behaviour under accident conditions using simplified neutronics models. This approach has limited validity for problems characterized by a strong coupling between neutronics and fluid dynamics in the primary circuit. The interdependence is not modelled properly, or very conservative assumptions would lead to unrealistic accident conditions. At present, the following accident conditions are under discussion which need analysis by coupled codes:
• The local boron dilution accident in PWR, which was identified as a potential reactivity initiated accident even in shutdown conditions when all control rods are inserted.

• The cooldown transients with strongly negative moderator temperature reactivity coefficient (MTC) in PWR. The occurrence of a recriticality during cooldown and its consequences have to be analyzed. Such high values of MTC are obtained for increased high burnup fuel or for extended use of MOX fuel.

• The results of ATWS analyses are strongly affected by feedback reactivity coefficients. The uncertainties of inherent feedback determining power production and consequently pressure increase can be strongly reduced by applying 3D neutronics models. The spatial effects are emphasized if partial failure of control rod insertion is postulated.

• The BWR instability in plant conditions beyond the stability threshold

Only the application of coupled codes will allow realistic simulations for these accident conditions.

3. General Considerations for Coupling

The basis of coupling are the available codes. The ATHLET code /3/ was developed by GRS for global plant transient and accident analysis. It is comparable to other system codes like RELAP or CATHARE. Furthermore, the 3D neutronics code QUABOX/CUBBOX was developed by GRS. It is based on an efficient coarse mesh method for solving transient two energy-group neutron diffusion equations for rectangular fuel assemblies typical for BWRs and PWRs. This paper is focussed on this code though several other 3D neutronics codes were coupled with ATHLET as well, like BIPR-8 from Kurchatov Institute, DYN3D from FZR Rossendorf and KIKO-3D from KFKI.
Starting from these codes different approaches of coupling seem to be reasonable:

1. Coupling of 3D neutronics models to the system code which models completely the thermal-hydraulics in the primary circuit including the core region.

2. Coupling of 3D reactor core models describing neutronics and thermal-hydraulics in the core region to the system code which models only the thermal-hydraulics in the primary circuit excluding the core region.

The pro's and con's are discussed in /4/. We prefer the first approach, mainly because it guarantees a consistent solution of thermal-hydraulics in the primary circuit. Therefore, numerical problems will be avoided caused by fast pressure disturbances, by transport of step changes in enthalpy or even in cases when mass flow reversal occurs during the transient. A general aspect is also that the numerical solution methods in the fluid dynamic code and the neutronics code are kept because they were optimized during their stages of development. It would be unreasonable to define a single matrix equation for the whole system and to solve it numerically. If different time-integration methods are used in different partial models, it is necessary to develop an efficient time-synchronization scheme.

It must be emphasized that in this manner all model features of the independent codes will be kept.

4. Implementation of Coupling

In the system code ATHLET an interface to 3D neutronics codes has been implemented which is applied to couple different neutronics codes.

The interface structure is designed in a very general way, corresponding to the main tasks of coupled calculations.
1. Reading input data consisting of the input data of independent codes and an additional part describing the relation between fluid dynamic channels and core loading.

2. Data exchange between fluid dynamic system code and neutronics, namely the transfer of power density distribution and feedback parameters between models.

3. Controlling for static solutions the iterations between fluid dynamics and neutronics to obtain steady-state conditions.

4. Controlling for transient calculations the synchronization of time integration by determining time-step size and the sequence of calculational steps.

It is supposed that the neutronics code is adjusted to these main tasks. Practically, it has been found that this structure was available in all codes studied.

From the programming aspect the interface is structured such that three levels exist.

This structure mainly contributes to limit the interference between fluid dynamic system code and neutronics code to a minimum. In order to maintain the independence of data between the codes, an additional set of arrays was defined for the interface level for each parameter to be exchanged. This allows that each code can keep its own indexing methods. The mapping of values is performed within the interface.

The first level consists only of general calls from ATHLET for main programme functions of the interface. In this layer all programme control is performed by ATHLET. It also includes the data exchange between ATHLET arrays and the interface arrays as explained above.

The second level consists of subroutine calls for the interface subroutines of the specific neutronics models. On this level, it is decided which neutronics model will be used.
The third level is specific for each neutronics model and necessary adaptations can be implemented. This level contains also the data exchange between interface arrays and data sets of neutronic models.

The main physical parameters, which must be exchanged between fluid dynamics and neutronics, are:

- The power density distribution. It is the result of the neutronics calculation and must be transferred to the fluid dynamics.

- The distributions for fuel temperature, coolant density and coolant temperature as well as the boron concentration. These parameters are the result of the fluid dynamic model including the boron transport model and must be transferred to the neutronics.

A specific feature of the implementation is the high flexibility in defining the mapping between fuel assemblies and flow channels. This mapping can be defined by input data. It optionally allows a grouping so that several fuel assemblies correspond to an average flow channel, or a 1 to 1 correlation that each single fuel assembly corresponds to a separate flow channel.

Also the axial meshes for neutronics and fluid dynamics can be defined independently. The necessary data transfer and also the interpolation for the axial direction is managed within the interface.

For static calculations it is only necessary to control the iterations between fluid dynamics and neutronics in an appropriate way. That is to determine the accuracy of power density distribution calculated in neutronics before starting the next fluid dynamic solution.

Attention must be paid to the time integration, because a strategy must be implemented, which considers all possible relations between time-step size in fluid dynamics and neutronics. The time-step size in fluid dynamics may be larger than, equal to or smaller
than the optimum time-step size in neutronics. Dependent on this relation the number of
time-steps and the time-step size itself must be chosen for neutronics.

In summary, the coupling of ATHLET with 3D neutronics code QUABOX/CUBBOX is
categorized as follows:

- Both codes keep their independent capabilities.

- The coolant flow in the primary circuit including coolant flow through the reactor
core is modelled by ATHLET, and the set of fluid dynamic equations is solved con-
sistently by the efficient numerical methods including time-step control based on ac-
curacy criteria.

- The time-integration of 3D neutronics is performed by the implicit matrix splitting
method used in QUABOX/CUBBOX.

- The time-integration of fluid dynamics and neutronics is performed with different
time-step sizes which are synchronized appropriately. Time-step control is defined by
fluid dynamics.

- The mapping of fuel assemblies and flow channels can be defined by input, as well as
the axial mesh for neutronics and fluid dynamics can be defined independently.

5. Experience from Applications

The coupled code ATHLET and QUABOX/CUBBOX was applied for various test
problems successfully. Typical examples are transients like an unintended control rod
withdrawal, an ATWS transient „loss-of-heat sink“, or the dilution of boron concentra-
tion by injection of demineralized water. All these problems were solved for a simpli-
fied PWR model. No systematic studies have been performed up to now with respect to
the number or mapping of flow channels or the axial nodalization. Transient calcula-
tions for realistic problems have been performed for RBMK reactors, where the full core was modelled in neutronics calculations. This really demanding problem demonstrates that computing times on fast modern workstations are not limiting for such calculations.

In the future, it is planned to validate the coupled codes by analysis of plant transient data or by code comparisons within benchmark problems. It is appreciated that an activity was started by OECD to define a relevant benchmark problem for coupled codes /5/.

6. Summary

For the realistic simulation of specific accident conditions the system code ATHLET was coupled with 3D neutronics models. The coupling approach was described, and the specific features of the coupled code system ATHLET and QUABOX/CUBBOX were presented. The implemented interface for neutronics codes is flexible enough to allow other 3D neutronics models to be coupled. Experience from first applications has confirmed that computing times are not limiting for such calculations. More studies on the effect of nodalization in fluid dynamics and neutronics are needed. In the future, the validation of these coupled codes will be continued by analysis of plant transient data and by code comparison within benchmark problems.

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Interfacing High-Fidelity Core Neutronics Models to Whole Plant Models

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Abstract

Until recently available computer power dictated that whole-plant models of nuclear power stations have typically employed simple models of the reactor core which can not match the fidelity of safety-qualified 2-group, 3D neutronics models. As a result the treatment of situations involving strong coupling between the core and the rest of the plant has inevitably been somewhat approximate, requiring conservative modelling assumptions, or manual iteration between cases, to bound worse case scenarios. Such techniques not only place heavy demands on the engineers involved, they may also result in potentially unnecessary operational constraints. Hardware is today no longer the limiting factor, but the cost of developing and validating high-quality software is now such that it appears attractive to build new systems with a wider simulation scope by using existing stand-alone codes as sub-components. This is not always as straightforward as it might at first appear. This paper illustrates some of the pitfalls, and discusses more sophisticated and robust strategies.

1. Introduction

The control systems of nuclear power plants are designed to achieve a degree of decoupling between the reactor and the rest of the plant. In many scenarios, therefore, we find it possible to model the behaviour of the whole plant with models that have fairly simple representations of the reactor core, and, conversely, to run detailed models of the reactor core with fairly simple external boundary conditions (perhaps derived from the whole-plant models).

There are, however, times when we would like to represent with a higher degree of accuracy those situations where the coupling between the core and the rest of the plant is not so easy to ignore. For example:
• analysis of operational transients which, though small, may, because of the frequency with which they occur, cause fuel fatigue;
• monitoring and fine-tuning of plant performance during normal operation;
• analysis of certain faults which produce strong interactions between core, boiler and the rest of the plant;
• building training simulators which will represent plant behaviour with an adequate degree of fidelity during fault situations - or even in normal operational conditions where precise representation of behaviour may be important.

These problems can only be tackled using simulations with plant-wide scope, but which must also deliver the degree of modelling fidelity which is now only found in a number of separate codes. Since it is now prohibitively expensive to develop and validate new software systems of the required scope and sophistication, reuse of existing qualified models as sub components in large systems appears highly attractive. Many papers have appeared recently describing various couplings (e.g. ref. 1 and several papers in ref. 2.)

This paper discusses couplings with PANTHER using several detailed examples. Other couplings, not discussed here, have also been made (e.g to RELAP5 & VIPRE). The history of this work is a movement from specific to more general methods of building such links.

2. Current Status

Three topics will be described in detail in order to illustrate the pitfalls with various code coupling techniques:

• the coupling of PANTHER to the SCORPIO system, for use in on-line core-surveillance and operational planning at Sizewell B;
• coupling PANTHER to the MACE code;
• generalising methods of coupling PANTHER to external codes.

2.1 Panther

PANTHER has a central role in this paper, and some understanding of its capabilities is essential. I shall not try to describe all its features here, but only those which are relevant to the discussion. The code itself, and the validation, have been described in a number of published documents [ref 3].

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It is now used by Nuclear Electric as a front-line code for modelling the core neutronics and thermal-hydraulics of both Advanced Gas Cooled Reactors (AGR's) and Pressurized Water Reactors (PWR's), covering:

- safety case analysis;
- fuel cycle design;
- operational monitoring (including safety-case compliance).

Other organisations have also used PANTHER to model a range of other reactor types, including MAGNOX, VVER, RBMK, and HTR.)

The adaptability of the package arises from a number of features, including:

- a highly modular structure, which supports both flexibility and maintainability;
- a very general data management scheme, using a single repository (the "Internal Data Base" or IDB) which holds the PANTHER data structures used during the execution of a physical model as IDB "records", all fully documented for the user and all, in principle, accessible to any external process via simple mechanisms;
- a high level "task description" language, which is used to configure all aspects of the functionality and data management. Frequently required sub-tasks can be encapsulated in parameterized "macros" for easy and secure reuse of validated procedures. Since macro definitions are stored as "records" on the IDB, they are capable of being manipulated in the same way as other data structures, and in particular can be constructed and transmitted to PANTHER by external processes.

In addition, built-in TCP/IP communications, combined with the above features, makes Panther a good system in which to investigate the issues of coupling large simulation packages, because many different types of interface behaviour can be provided merely by changing the "task description. No source code alterations are required: all these capabilities are supplied using a standard production load-module subject to strict quality assurance. Furthermore, standard PANTHER data viewing facilities can be employed to examine and verify in detail the interactions between PANTHER and other systems. Such considerations are important when qualifying an application for use in a safety-related role.

In order to illustrate how this works in practice, I shall outline how PANTHER can be set up to act as an event-driven calculation server, capable of being configured to supply results to a variety of external processes. In this example the top level of the task
description is used to specify "session management", i.e. initiating a communications link and synchronising the exchanging messages according to an agreed protocol. (Some non-essential details, e.g. error handling, have been omitted for clarity.) Anyone who has every programmed a "server" process in any language will recognised the classic design pattern.

[===============================================================================]
( Set up a "port" to which external processes can request connection )
BIND TO LOCAL PORT 5432 AS "PANTHER_PORT" (Reserve for our use )
ACCEPT CONNECTIONS ON "PANTHER_PORT" FROM TASK "REMOTE" (Wait for call )
( Return from above command only when connection request is received )
[===============================================================================]

[===============================================================================]
( Enter an infinite loop, responding to client commands )

DO FOREVER  (Actually until the REMOTE task sends shutdown signal )
RECEIVE FROM TASK "REMOTE" {Standby for message, returning when }  
{ all incoming data has been copied to IDS as records. One of these }  
{ records MUST be the definition of a macro called "WorkRequest" }  

EXECUTE MACRO "WorkRequest" {Carry out requested work from REMOTE }  

{Protocol requirements: }  
{ - The data sent back to the REMOTE task as the message response }  
{ must be specified in the macro definitions, as with: }  
{ "SEND TO TASK "REMOTE" RECORDS WITH <data-spec>" }  
{ - EVERY request must also be acknowledged with the following }  
{ synchronisation flag, even if no other data is being returned. }  
SEND TO TASK "REMOTE" END OF MESSAGE

{Protocol requirement: REMOTE sets flag REMOTE_FINISHED when done}  
LEAVE IF $REMOTE_FINISHED EQ 1   {REMOTE has set the "exit" flag }  
LEAVE IF $ERROR_COUNT GT 1     {Error found - exit }  
ENDDO

CLOSE REMOTE                   {shutdown external connection}  
[===============================================================================]

This configuration imposes a number of protocol requirements on the remote task, in particular, the message transmitted from the external process at each cycle must always contain the definition of a macro called "WorkRequest", which is used to specify the action required from PANTHER as a response to the message.

In principle, the WorkRequest macro can contain any legal PANTHER commands, so the remote process could specify the PANTHER task behaviour in a completely dynamic manner. In practice it is easier to work within a more constrained framework, and one of the first requests sent by the external process will normally contain a command such
as:

RETRIEVE FROM FILE "APPLICATION_MACRO_LIBRARY"

This command loads a library of macros definitions from disk, which can be used to specify a standard set of messages to which the PANTHER server will give a known response. Subsequent WorkRequest macros might then, for example, contain requests such as:

EXECUTE MACRO "ESTABLISH_INITIAL_MODEL" STATE_FILE="<file-spec>"

or:

EXECUTE MACRO "DO_STATE_POINT" BANK_Z=0.78 POWER=100.0

or

EXECUTE MACRO "TIME_UPDATE" DT=1.2-3 BANK_Z=0.5 POWER=95.0

With this type of approach the WorkRequest macros therefore contain only the name of an incoming "event", plus, perhaps, some event data supplied as macro parameters. (Other data can also accompany the message as PANTHER "records" which are copied directly to the IDB, from where they can be accessed by the PANTHER calculation.) The action which occurs as the response to the event is specified by the macro library which is selected for use. A different library will respond to a different set of incoming events in a different way.

In practice many applications share common patterns of behaviour, so it is possible to reuse the same framework of messages. For example, many reactor modelling applications will probably need to initialise, establish steady-states, and have a procedure moving the model state forward through time. In some cases we might, for example, decide to interpret the "time_update" request as an irradiation update, in others a xenon state update. The external process will generally expect a reply to its message, usually in the form of data, but perhaps just a synchronisation event which initiates further action in the remote task.

Data is returned to the remote application, according to its requirements, by including in the macro definitions one or more "SEND" commands, which transmit specified PANTHER data structures through the TCP/IP socket. In practice the required responses for a given application tend to fall into a small number of standard patterns, so we encode these as a lower level of macro definitions. In the simplest case each of the event-handling macros might call the same procedure (which might be specified, or redefined, at run-time):

EXECUTE MACRO "STANDARD_REPLY"

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The detailed format of all messages conforms to a simple, but PANTHER specific structure. The external process uses a standard utility library to pack and unpack message contents, so the work of creating a PANTHER interface to an application is the encapsulation of a number of such calls within a few subroutines to create a simple view of the core model. We find it convenient to create a form of "remote procedure call", my mapping macros defined in the application macro library to local subroutine names.

SUBROUTINE PAN_START("<application_macro_file_name>")
SUBROUTINE PAN_MODEL_DEF("<model definition file>")
SUBROUTINE PAN_STREAM(BANK2, POWER, <returned data>)
SUBROUTINE PAN_UPDATE(DT, BANK2, POWER, <returned data>)

It is possible to define both synchronous interactions (where the call does not return until PANTHER issues its response) and asynchronous interactions (where the initiating call returns as soon as the message has been sent to PANTHER, and where the response must be gathered by a later subroutine call). We shall see later that it is possible to extend these concepts in order to build highly object oriented interfaces to PANTHER models.

2.2 PANTHER and SCORPIO

SCORPIO is a core monitoring system developed at the Halden Project [ref 4] which has been implemented at Sizewell B [ref 5]. During the first fuel cycle SCORPIO employed the standard "CYGNUS" core model supplied by Halden. However, prior to the second fuel cycle, the CYGNUS model was replaced with PANTHER. The incentive for the change was two-fold:

- a reduction in the work necessary at each fuel reload to re-qualify the core model (essentially down to zero, since a standard Panther model will in any case be qualified for the new fuel cycle);
- in order to support "Automatic Frequency Regulation Operation", improved accuracy and reliability of predictions were essential.

The substitution process is straightforward, because SCORPIO passes a small set of data to the core model, and it expects from it a relatively simple repertoire of behaviour (start-up, establish initial state, update state, shutdown, using as data rod positions, reactor power, and time-step data). In reply, SCORPIO expects to receive a small range of scalar parameters (e.g. boron concentration) and a 3D rating distribution. Almost inevitably, the physical units and mesh layouts used by PANTHER and CYGNUS are different - and SCORPIO expect the CYGNUS conventions. However, it proved possible
to build a small module in SCORPIO, with the same interface as CYGNUS, which understood the SCORPIO conventions, and was also able to ask PANTHER about its conventions at run time. Hence we can create a translation scheme which adjusts itself to the requirements of the particular PANTHER model currently in use. All that remains, using the techniques described in the last section, is to define a small set of PANTHER message-handing macros which are able to meet the requirements of SCORPIO.

In general, there will always be a need for translation to ensure that a message transmitted from one process is interpreted with the same meaning by another process. When the messages being passed talk about real world entities it is relatively straightforward to find a context in which to agree a common meaning. Everyone can easily agree about what they mean by a 'control rod bank' (though perhaps not about the units in which bank insertion is measured). Messages which talk about actions to be performed can be sensibly referred back to real actions in the world (e.g. move a rod bank to a certain location). However, when handling information about spacial distributed quantities, such as neutron flux, the situation become more complex, because there are many ways of specifying a discrete approximation to a continuous real world distribution. In the absence of prior conventions it is most unlikely that there will be any uniformity of representation. Interpolation from one representation to another is easy using standard methods, but only after external guidance has provided the understanding of how each representation relates to the real world in order to understand how they relate to each other.

2.3 PANTHER and MACE

MACE [ref 6] is a program used to simulate reactor circuit problems for AGR’s and MAGNOX reactors. In effect, it is the gas reactor equivalent of the the RELAP5 code, being designed to model situations involving breaches of the primary circuit. Because of the similarities in modelling techniques and the software architecture, much of the discussion below applies equally well to RELAP5.

Ultimately, the purpose of most fault analysis is to demonstrate the integrity of the fuel within the core. This is outside the direct capability of MACE, but MACE simulations of a fault transients are often used to specify the flow and core inlet temperature boundary conditions for a PANTHER treatment of the same scenario. However, MACE requires as one if its boundary condition the heat distribution within the reactor core, for which a PANTHER calculation is needed. At present the only way to perform the required iteration is by manual transfer of tables of the way the boundary conditions vary with time for the complete transient, which is inefficient both in computer time and
engineers' time. It would be more desirable to exchange boundary conditions at each time point, during the progress of the simulation.

The interfacing issues are somewhat more complex than those encountered with SCORPIO, since both PANTHER and MACE are general purpose codes, each capable of being configured to represent a variety of fault scenarios, in a various degrees of modelling approximations. Since we do not wish to rebuild the software each time we look at a new scenario or change the model, we must design a reconfigurable interface.

What sort information must be supplied to configure a link? PANTHER simulations generally talk to the user in terms of objects with which he is familiar (fuel assemblies, pins, control rods etc.) and details of the numerical model (e.g. detailed mesh layouts) are usually hidden from the user. MACE, however, takes a different approach: the problem description supplied by the user is already in terms of abstract modelling entities such as Volumes, Junctions, Structures etc.. The relationship of such abstract entities to real world entities is not used by MACE. This means, for example, that PANTHER might expect to be told the flow and inlet temperature of a fuel channel, while MACE knows how to calculates flow and temperature at, say, Junction #75. Going in the other direction, PANTHER predicts the axial heat production along, say, channel "M:25", while MACE needs to be told the heat sources associated with the Structures having common surfaces with a specified series of Volumes. The two codes therefore have fundamentally different semantics, and some serious translation issues must be addressed.

Specification of variable mappings directly from attributes of MACE entities (e.g. Junction #75 flow) is perfectly feasible but has disadvantages. In particular such entities have no essential relationship with the real world object they model - it would, for example, be possible to get exactly the same MACE model with a different node numbering scheme. The specification therefore becomes dependent on the exact form of the MACE problem specification, and requires the user to interpret how this relates to the real world.

We were able to avoid some of these problems by using a facility within MACE that allows us to attach arbitrary labels to entities within the model. These we can employ in order to create a mapping back to the real world objects with which they should be associated (this is, in any case, their normal function in documenting the MACE model). For example Junction #75 might be labelled "M:25_channel_inlet". It is then possible to map PANTHER data structures to MACE data structures, in an understandable and robust manner using the common reference to the same real world entities, as in:


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The detailed mechanics of reading and writing MACE internal data structures at appropriate points during execution are essentially the same as those described for RELAP5 by a number of other authors (e.g. in ref 2), and we will not discuss them further here.

Mapping of data structures is, however, not the entire story: we also need to handle the dynamical behaviour. Since part of the justification of this work is the reuse of existing software with minimum modification, the behaviour of MACE is not changed in any way (other than by modification of boundary condition values), it is only asked to send a signal to the PANTHER interface when it enters one of its normal execution phases, which are:

- initialisation;
- read re-start file (on user request)
- perform model update (exchanging boundary conditions with PANTHER);
- back-step (if model update generated unstable numerical conditions);
- write re-start file (at user specified intervals)
- close-down.

Note that MACE sometimes finds that it must back-step to a numerically stable point, so a corresponding back-stepping behaviour has therefore been defined in the PANTHER task. No other iteration of boundary conditions occurs, since the time steps would normally be small enough to render this unnecessary. However, the PANTHER architecture (which allows arbitrary numbers of model states to be held simultaneously on its IDB) would allow the specification of more complex global iteration schemes.

3. Towards Generic Components

The previous sections illustrate only two examples of coupling PANTHER to other systems, in order to illustrate issues of particular interest. However, other coupling also exist (e.g. PANTHER/RELAP and PANTHER/VIPRE), and more applications are under consideration - including the use of PANTHER in training simulators. It is clearly wasteful of effort to employ different techniques each time such a project is undertaken, so recent work has concentrated on defining a generic framework by which external systems may interact with a PANTHER simulation. This strategy defines the essential features of a generic reactor model as they should appear to external system, and then builds an interface which gives PANTHER the ability to behave according to this specification.
The major technical problem when exchanging information between separate simulation models is ensuring that the interpretation of the message as sent is the same as the interpretation as received. The simplest strategies for translation correspond to using a "dictionary" to provide mappings between words in the two languages. Such approaches tend to fail because the meaning of words is nearly always context dependent. Consider for example, a quantity labelled "gas outlet temperature". In many situations it might not matter whether we mean the real gas temperature, or the thermocouple indication of this quantity; however, the reactor trips on the indication, not the real gas temperature, and the two values may differ significantly in a simulation of a fast fault scenario. "Phrase book" strategies are somewhat better, because they are based around particular scenarios which define a restricted pattern of interactions. This approach may give us core models which are valid only for a limited range situations; once we depart from the original scenario, we loose the context which is used to resolves ambiguous meanings. The PANTHER/SCORPIO and PANTHER/MACE links are examples of phrase-book translation strategies. They work because the scenarios being executed define a limited set of uses for any message, and the correct translations of all messages have been confirmed within the particular context. If we wish to define a more general interface to a core model "component", which can be employed over a wider range of applications, it must therefore necessarily involve a more extensive model of the "universe of discourse" in which the meaning of any valid message is defined by the complete set of uses to which we can put the received information.

The limitations of previous techniques became apparent during the design a graphical user interface to PANTHER, which would be capable of dealing with both AGR and PWR reactors. PANTHER models of these reactor types have both similarities and differences. Some of the differences are merely in labelling ("assemblies" in one context are called "stringers" in another etc.), or in the conventional units used to quantify model parameters, but some relate to real modelling differences. For example, AGRs always have at least one control rod bank where the rods move independently to control radial power shape - a process which must be modelled. Furthermore, Nuclear Electric's five AGRs are built to four different designs, with four different safety cases - each requiring somewhat different parameters to be calculated during fuel cycle design. The users expect the GUI to adjust to their normal view of the problem domain, and this view differs for each reactor type in ways more fundamental than mere changes in screen layout: we can express this in in a more formal way by saying that the behaviour must correspond to their particular requirements model.

From one point of view a PANTHER model presented using the techniques of section 2.1 can be regarded as an encapsulated "object" - it offers a number of external interfaces, and has many internal states, which are changed by invoking the interfaces.
The behaviour of this object (i.e. the interaction of interface invocations with internal states) could be precisely documented using, for example, a large state transition diagram. Such an approach is entirely feasible where the repertoire of required behaviour is as simple as in the PANTHER/MACE link, but quickly becomes complex and difficult to understand as more interfaces and states need to be added in order to generalise the model. Furthermore, it is still be highly oriented toward the "PANTHER" view of the world, not the users' view. What we need is the ability to reconfigure the interface at run time to present a pattern of behaviour which can be interpreted consistently within an appropriate and understandable world model.

This requirement is achievable using object-oriented methods. The most powerful aspect of these techniques is an ability to represent complex abstractions, with a formal mechanism for specialising the abstract model to particular examples. We can, therefore, find ways of encompassing different views of a reactor model within the same global representation. Furthermore, we can readily implement our abstract model in, for example, C++ classes, in a way that allows us to make the required specialisations at run time (by selecting for instantiation particular object types) and so actually produce the particular behaviour required for the current role in which PANTHER is being used. The original abstract model is a formal specification of *potential* interface behaviours, the addition of choices made during specialisation produce a formal specification of *actual* interface behaviour.

It is relatively easy to build this type of object-oriented "wrapper" module for PANTHER. We are able to access easily almost any data structure, and configure many different types of behaviour. Furthermore, PANTHER already includes explicit descriptions of many real-world entities, such as fuel assemblies and core-locations and the relationships between them, which can be exported to the wrapper for use in run-time configuration. The current implementation of this interface is able to support tasks connected with the design of fuel-cycles for both AGRs and PWRs. Part of the run-time specialisation comes from the application, part comes from the PANTHER case. In principle, for example, the application may say "I want to use axial offsets", but if we have made available a 2D reactor model, the reply will be "No - I can not do that" (or it might be "wait while I load the 3D model"). The ability to answer questions about capabilities is a valuable facility, which is easy to provide with this type of approach.

The same techniques could be applied to systems such as MACE or RELAP5, but would require more work, since we would need to supply more information to specify how entities and attributes in the MACE model relate to a higher-level view. There are, however, some good reasons why one might wish to undertake such work. Although the MACE model might be expressed in terms of *Volumes, Junctions* etc. it is really a
representation of objects such as fuel channels, boilers, turbines, pumps and so on (and this is precisely the sort of information we supply if, for example we build a graphical interface to RELAP or MACE). Some of these objects (e.g. channels) are the same objects we represent in the PANTHER simulation, so we ought to be able to simplify our communications problems if, where the high-level MACE interface model overlaps with the PANTHER interface model, they are the same model. This is rather more than simply compiling a common data dictionary and common data representations - it is the network of relationships associated with an item of data, and its potential uses that give it a unique meaning; we can only ensure a common interpretation by requiring both parties to conform to a common model that encompasses all such information. Such a model is also rather more than a means of connecting two specific codes such as PANTHER and MACE; it will necessarily focus on the environment in which a component must operate, and therefore takes on the role of a requirements model which becomes independent of either code. We find, therefore, that we can built a framework which facilitates one implementation of a component to be substituted by another. Differences in capability are dealt with during the initialisation of the "wrapper" modules, which declare which of the standard services they are able to offer to other components. (Other conditions must also be met, of course, such as use of common communications protocols.)

The full details of interface models for system such as PANTHER and MACE are too complex to present here. However, we can illustrate the essential workings of this type approach with a simple, but realistic example. Consider the problem of converting a data item expressing by code A in one physical unit, to be understood by code B which uses another unit. The two codes need to negotiate a unit conversion. We will assume they do this using an implementation based on the data model of figure 1 (which is expressed in Schlaer/Mellor style - see ref 7), which describes the types of object that can exist in the system. Initially, the implementation knows nothing about any real units, it must be primed by instantiating a few objects supplying some basic unit definitions, such as meter, second, etc.. On creating these instances we say how these units relate to some universal standard, such as the Standard International. (e.g. when we define foot we say "multiply by 0.3048 to get to the standard length unit"). Units such as "litre" can then be defined in terms of powers of the basic units - by instantiating further objects holding the relationships required by the model.

Suppose now that the two codes, A and B, agree to exchange a particular value, but A declares that it is measured in litres, and B in cubic feet. The unit conversion module would look up litre, and find (a) it was a volume unit and (b) the conversion factor to the standard scale would be 0.001. It also be presented with a unit definition "ft**3", which will be found to have the same dimensionality as litres (so the proposed
conversion is legal), and then the conversion factor to the standard scale would be calculated as \((0.3048 \times 3)\). Now a “Conversion” object instance is created with a translation factor attribute set to 2.8317 \(\text{ft}^3/\text{litre}\). A reference to this object can now be passed to both A and B, which they can use to invoke the negotiated translation service in all future communications involving the value they wish to exchange. This model can also support the capability (via Unit Class objects) for either party to give a global declaration such as “every time I talk about volumes they will be measured in litres” and then simply label its interface parameters as “Volumes”, “Pressures” etc., rather than with individual unit definitions. (A service conforming to this design is in fact included within PANTHER to facilitate communications with the external world.)

<table>
<thead>
<tr>
<th>Definition</th>
<th>for C</th>
<th>Unit</th>
<th>from C</th>
<th>Conversion</th>
</tr>
</thead>
<tbody>
<tr>
<td>*UnitFor</td>
<td>C</td>
<td>*UnitName</td>
<td>C</td>
<td>*UnitFrom</td>
</tr>
<tr>
<td>*UnitOn</td>
<td></td>
<td>FactorToSI</td>
<td></td>
<td>*UnitTo</td>
</tr>
<tr>
<td>PowerOff</td>
<td></td>
<td></td>
<td></td>
<td>ConvertFactor</td>
</tr>
</tbody>
</table>

When a unit is not defined in terms of other units, it MUST have its SI conversion factor defined on creation, otherwise this will be derived from the definition.

<table>
<thead>
<tr>
<th>Unit Class</th>
<th>i.e. Energy, Power, Pressure, etc.</th>
</tr>
</thead>
<tbody>
<tr>
<td>*ClassName</td>
<td></td>
</tr>
<tr>
<td>MassPower</td>
<td></td>
</tr>
<tr>
<td>LengthPower</td>
<td></td>
</tr>
<tr>
<td>TimePower</td>
<td></td>
</tr>
</tbody>
</table>

Figure 1: A Model for A Unit Conversion Service

An essentially similar technique can also be used to negotiate interpolation services, by providing a standard model of different ways in which spacially distributed quantities can be represented, to which each code can compare itself. Although this is more complex, it is possible to describe the most likely representations, and hence translations, with a relatively compact abstract model. Once two codes have each declared their chosen form of representation, the interpolation service will return a “handle” to both codes which they can then use to invoke an appropriate conversion operation.

It has not escaped my notice that these techniques also provide a natural route to a distributed implementation of coupled models by the use of, for example, CORBA technology (the Common Object Broker Request Architecture). Services such as unit conversion and interpolation would, in this architecture, be accessed as a common service via the Object Request Broker, along with access to the physics codes.
4. Summary

There is a growing requirement to incorporate pre-existing fully validated high fidelity models as components into large integrated simulation systems. However, such models often vary widely in the way they view the world, and sometimes these differences are subtle and easy to miss. In order to gain the benefits of using an existing software package we must avoid modifications, so some form of translation during message transfer is usually required. In order to avoid errors it is desirable that all differences in world view should be made explicit by reference to standard information models, which can be used to describe the interface behaviour of all participating components, and that translation between data representations should then be negotiated automatically. Using modern software techniques this appears to be a feasible option. The apparent complexities of this approach are already inherent in the nature of the problem, and not an artifact of the techniques.

References


Neutronic Aspects of the THOR Core Dynamic Model for Training Simulators

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1. INTRODUCTION

The THOR advanced thermal hydraulic and neutronics model simulates the NSSS dynamic behavior of BWR and PWR plants in real time, for application in full scope training simulators. THOR is tailored for implementation in the OpenSim NT simulator executive environment on Windows NT platforms. This paper focuses on the neutronic aspects of the THOR model.

The nodal 1.5 group approximation of the diffusion equation is based on the neutronics model of RAMONA-3, a well-qualified engineering design code. Although, in engineering analysis applications, the model is normally applied with a nodal mesh of 6 inch sides, the real time applications in training simulators call for larger nodes. All control rods and detectors are explicitly represented in the model.

Nuclear cross-section data are generated with methods consistent to those applied in core design and supervision. Polynomial expansions of the data in coolant density and fuel temperature are provided to assure fast execution as well as robustness in the whole range of core transient scenarios. The reactivity influence of control rods, Xe, Sm and Boron is modeled through corrective terms to the two-group basic macroscopic cross sections. Systematic methods are applied for cycle-wise updating of the core model.

The requirement for real time simulation imposes restrictions in the number of nodes in the neutronics as well as thermal hydraulics. A "mapping" procedure is applied to distribute the coarser mesh T/H quantities (void, temperature) into the finer neutronic nodes by the application of basic physical principles.

Results are presented to illustrate the effect of fast power transients as well as the model behavior under extreme transient conditions. The performance of THOR on a Pentium PC computer is presented
2. CORE DESCRIPTION

The 3D distribution of power generation, fuel temperature as well as coolant density and temperature in the core is dynamically calculated by THOR. The distributions are discretized into nodes, typically designed as follows:

Neutronics (power, neutron flux): For PWRs, one fuel assembly constitutes one radial node, with an axial discretization into 12 nodes; typically altogether ~2200 nodes.

For BWRs, four fuel assemblies are radially grouped into one radial node, with an axial discretization into 12 nodes, i.e. also about 2200 nodes.

Thermohydraulics (coolant cond.): The coolant flow is modelled in a number of vertical parallel channels. Several fuel bundles are lumped into each hydraulic channel: for BWRs, typically the core is represented by 5 active and 1 bypass channel; PWRs are usually represented by 1 active and 1 bypass channel.

All control rods are individually described. The response on neutron detectors (source, intermediate and power range, fixed and movable) is being simulated. Ex-core detectors and thermocouples are included in PWR models.

3. NEUTRONICS MODEL

3.1 Nodal Methods

The THOR core model is based on the 3D neutronics model used in the RAMONA-3 system transient code (Wulff, 1984). The RAMONA-3 code was developed as an engineering analysis code and has been used extensively to calculate the dynamic response of BWRs during normal and abnormal plant transients.

Two-group diffusion theory forms the basis for the neutron kinetics of THOR:

\[
\frac{1}{\nu_1} \frac{\partial \phi_1}{\partial t} = \nabla \cdot D_1 \nabla \phi_1 - \Sigma_{\text{m}} \phi_1 + (1 - \beta)(\nu_1 \Sigma_{f1} \phi_1 + \nu_2 \Sigma_{f2} \phi_2) + \sum_m \lambda_m C_m
\]

(1)

\[
\frac{1}{\nu_2} \frac{\partial \phi_2}{\partial t} = \nabla \cdot D_2 \nabla \phi_2 - \Sigma_{\text{m}} \phi_2 + \Sigma_{\text{c}} \phi_1
\]

(2)

\[
\frac{\partial C_m}{\partial t} = \beta_m (\nu_1 \Sigma_{f1} \phi_1 + \nu_2 \Sigma_{f2} \phi_2) - \lambda_m C_m, \quad m = 1, \ldots, 6
\]

(3)
The approximations of THOR are based on PRESTO nodal methods (Børresen, 1983):

- Finite difference approximation of the spatial dependence and application in a coarse mesh geometry.

- Replacement of the thermal flux divergence term $\nabla \cdot D_i \nabla \varphi_i$ by an approximate expression $\Lambda_2$ based on the asymptotic thermal flux in the node midpoint and corresponding values in the six neighboring nodes.

- An empirical expression is applied for the node average flux calculation from the interface fluxes between the node considered and its neighbors:

$$\bar{\varphi}_i = b \varphi_i + \frac{1 - b}{b} \sum_{j=1}^6 \varphi_j^i$$  \hspace{1cm} (4)

where $\varphi_j^i$ is the interface fast flux between node (i) and its neighbors (j), and $b$ is an empirical constant.

- Approximation of expression $\frac{2D_i D_j}{D_i + D_j}$ by the product $\sqrt{D_i} \sqrt{D_j}$

With the approximation of the thermal leakage and backward differencing of the time derivatives, Eqs. (1) through (3) become:

$$\frac{1}{\nu_i \Delta t} (\varphi_i^{n+1} - \varphi_i^n) = \nabla \cdot D_i \nabla \varphi_i^{n+1} - \sum_{a=1}^4 \varphi_i^{n+1}$$

$$+ (1 - \beta) \left( \nu_1 \Sigma_{j=1}^{n+1} \varphi_1^{n+1} + \nu_2 \Sigma_{j=2}^{n+1} \varphi_2^{n+1} \right) + \sum_m \lambda_m C_m^{n+1}$$  \hspace{1cm} (5)

$$\frac{1}{\nu_2 \Delta t} (\varphi_2^{n+1} - \varphi_2^n) = \Lambda_2^{n+1} - \sum_{a=1}^4 \varphi_2^{n+1} + \sum_{a=1}^4 \varphi_1^{n+1}$$  \hspace{1cm} (6)

$$\frac{1}{\Delta t} (C_m^{n+1} - C_m^n) = \beta_m (\nu_1 \Sigma_{j=1}^{n+1} \varphi_1^{n+1} + \nu_2 \Sigma_{j=2}^{n+1} \varphi_2^{n+1}) - \lambda_m C_m^{n+1}, \hspace{1cm} m = 1, \ldots, 6$$  \hspace{1cm} (7)

Introducing the expression for $\varphi_2^{n+1}$ from Eq. (6) in Eq. (7) and substituting the result in Eq. (5) gives

$$-\nabla \cdot D_i^{n+1} \nabla \varphi_i^{n+1} = A \varphi_i^{n+1} + B$$  \hspace{1cm} (8)

where $A$ and $B$ depend on the nuclear parameters and thermal leakage at time $n+1$ and on the fluxes and precursor concentrations at time $n$. 

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A closed form expression of the flux equation is derived by applying approximations (1), (3) and (4) and making the variable transformation:

\[ \psi_i^t = \phi_i^n \sqrt{D_i} \]  

(9)

The flux in node \( i \) is then expressed in terms of the fluxes in the six neighboring nodes \( j \) by the following equation:

\[ \psi_i^{n+1} = Q \sum_{j=1}^{6} \psi_j^{n+1} + T \]  

(10)

which is the flux equation solved at every time step in the simulation.

The reflector effect is accounted for by the additional fast flux leakage and slowing-down terms defined for the peripheral nodes. These albedo parameters are pre-calculated with a procedure employing a fine mesh diffusion solution.

3.2 Neutron Cross-Section Representation

During the dynamic simulation, the nuclear cross-sections are treated with a functional dependence on the following nodal properties:

- Coolant density
- Fuel temperature
- Control rod presence
- Boron concentration
- Transient xenon concentration
- Transient samarium concentration

The dependence on coolant density and fuel temperature is modeled using polynomials in density and linear variations of the square-root fuel temperatures. Polynomials in exposure and density history are used to model the burnup dependence.

3.3 Delayed Neutron Data

Delayed neutron fractions are functions of fuel design and burnup. For fast transients near prompt critical, the delayed neutron fraction may be an important factor that needs to be treated accurately. THOR's kinetics uses six delayed neutron groups and treats the delayed neutron data individually for each node.

3.4 Power Generation

The thermohydraulic calculations require the heat generation or power density in each computational cell. The power density at any point is the sum of two components:
Prompt fission heat: the amount of energy released promptly in the fission process. 

Decay heat: the amount of energy released by the decay of the fission products. It is delayed relative to the prompt fission heat and hence depends on the fission rate history.

The prompt power density at spatial location \( x \) and time \( t \) is related to the fission density \( F(x,t) \) by

\[
q_p(x,t) = Y(1 - \alpha_r) F(x,t) \tag{11}
\]

with
- \( Y \): total energy per fission (~200 Mev/fission)
- \( \alpha_r \): fraction of the total fission energy that is delayed

The amount of delayed energy released by the decay of the fission products following the shutdown of a reactor is fitted with a series of decay heat groups. Thus the model is similar to the handling of delayed neutrons. The concentration of decay heat precursors \( D_i \) in decay group \( i \) is expressed in each neutronic node by the following differential equation:

\[
\frac{dD_i}{dt} = \alpha_i \sum_{k=1}^2 \kappa_k \Sigma_{k} \varphi_k - \lambda_i D_i \tag{12}
\]

where \( \alpha_i \) represents the fraction of the total energy appearing as decay heat in decay group \( i \).

### 3.5 Power Deposition

THOR takes into account the fact that fission energy is deposited as thermal energy both inside the fuel pellet where the fission takes place and outside the pellet due to neutron and gamma attenuation. Also, the heat deposition location for prompt and delayed power may be different. The heat deposition outside the fuel is split between direct heating of the moderator in the active channels and direct heating in the bypass channel. In a BWR, direct heat deposition can be significant because it creates an almost instantaneous void reactivity feedback during a transient.

### 3.6 Transient Xenon and Samarium Calculation

Time-dependent nodal concentrations of I-135, Xe-135, Pm-149, and Sm-149 are calculated in THOR using local fission rates \( F \) and local thermal flux values \( \varphi_2 \). The differential equations are integrated numerically using an implicit time-integration scheme. Provision for a fast integration switch is included.
The following equations are solved at each time step:

\[ I^{(\text{new})} = \frac{I^{(\text{old})} + \gamma_1 \Delta t \cdot F^{(\text{old})}}{1 + \lambda_1 \Delta t} \]  
(13)

\[ X^{(\text{new})} = \frac{X^{(\text{old})} + \lambda_2 \Delta t \cdot I^{(\text{new})} + \gamma_2 \Delta t \cdot F^{(\text{old})}}{1 + \lambda_2 \Delta t + \sigma_x \Delta t} \]  
(14)

\[ P^{(\text{new})} = \frac{P^{(\text{old})} + \gamma_p \Delta t \cdot F^{(\text{old})}}{1 + \lambda_p \Delta t} \]  
(15)

\[ S^{(\text{new})} = \frac{S^{(\text{old})} + \lambda_s \Delta t \cdot P^{(\text{new})}}{1 + \sigma_s \Delta t} \]  
(16)

### 3.7 Detector Models

The in-core detector response is predicted as an average of the power in the surrounding nodes. The weighting is performed by means of geometrical weighting factors as well as detector constants. The detector constants are specified individually for each detector string and its surrounding channels. The normalization constant is treated differently for SRM, IRM, and LPRM detectors.

Top and bottom ex-core detectors signals are constructed based on weighted sums of the flux at the core boundary, taking into account changes in liquid density in the downcomer.

### 3.8 Numerical Integration

The neutronics are integrated by a backward time difference formula, which is stable for any time step, and with the accuracy given by the convergence criteria of the predictor-corrector method applied. The parameters in the flux equation are functions of the core parameters at time step \( n + 1 \) and therefore dependent on the flux at the same time through feedback (via void, fuel temperature, etc.). This feedback can be estimated with sufficient accuracy by using a linearly extrapolated value on the flux from time step \( n \) to \( n + 1 \). This is achieved by using a predictor-corrector integration scheme.

### 3.9 Cycle-wise Core Model Update

An important aspect of the core model of a training simulator is the ability to update the nuclear cross-section database to new cycles and new cores. The principle employed in THOR is to represent the cross-section data in analogy with the static simulators employed by the in-core fuel management group. Normally, 2-group macroscopic diffusion parameters are employed today in LWR core simulators.

Data processing tools are provided together with THOR, that reads the cross-section database as generated by the lattice code (e.g. HELIOS, CASMO, etc.) together with
core burnup and core state files (as generated by e.g. PRESTO, SIMULATE, etc.). The core update utility will first generate all cross-section based on the standard static simulator node size (6x nodes) and then condense the data and produce the polynomial fits and specific coefficients required by the THOR neutronics model.

4. VALIDATION

The THOR neutronics model originates from the PRESTO nodal model which has been in production use for in-core fuel management applications for many years (Børresen, 1981). The diffusion theory approximation of PRESTO has been validated both against reference calculations and reactor measurements such as BWR gamma scan data, TIP traces, cold critical measurements, etc.. As an example of benchmark calculation results, the 3D IAEA benchmark (Argonne, 1977) gave a standard deviation of 1.16% in the bundle power and a maximum deviation of 2.6% for 20 cm cubical nodes. Comparison with the Hatch gamma scan data (Shiraishi, 1978) produced an overall nodal standard deviation of 6.4% and about 2% for the bundle power comparison.

The neutron kinetics formulation originates from RAMONA and is also thoroughly validated. It has been used for transient applications over many years, including qualification against plant measurements such as:

- Peach Bottom Turbine Trip Tests (Moberg et al., 1981)
- SPERT Reactivity Insertion Accident Transients (ABB-CE, 1996)
- Brown's Ferry Half-ATWS Incident (Wulff, 1984)
- Ringhals Stability Benchmark (Lefvert, 1996)

5. TRANSIENT CALCULATIONS

5.1 Applicability

The THOR neutronics model is applicable to all transients listed in ANSI/ANS Standard 3.5-1985, Appendix B and performs in compliance with the standards. Examples of such transients where an accurate neutronic model are especially important are:

- Control rod ejection / control rod drop
- Control rod maneuvering, SRI
- Pressure transients / turbine trip, load rejection, MSIV closure
- Pump trips
- Feedwater controller failure / loss of feedwater heater
- Xenon transients
- ATWS

Examples of significant neutronic parameters explicitly calculated, that can be verified against start-up physics test data, are:

- core power distribution
- moderator temperature coefficient
- Doppler coefficient
- critical Boron concentration and Boron worth
5.2 Results for a Turbine Trip Transient

Calculated results for a turbine trip transient are presented in this section. The power plant is a typical GE BWR. The steady state condition preceding the transient is at end of core life conditions at 93% nuclear power. No control rods are initially present in the core.

Closure of the turbine stop valves causes a fast increase of the system pressure and rapid collapse of the void in the core. The void collapse gives rise to a power spike with the peak reaching about 400% nominal power. The reactor scrams on high flux after the peak power is reached. The behavior of steam dome pressure and reactor power is illustrated in Figure 1.

Figure 1. Power and Pressure Response

Figure 2 shows the change in axial power shape during this transient. The power is initiated slightly bottom peaked. When void collapses the power shifts toward the top of the core and as the control rod are inserted during the scram the power the power peaks even further to the top. After the power surge and insertion of all the control rods, the power shifts back to the bottom.
5.3 Performance

The performance of the THOR prototype are illustrated in Table 1 below for some limiting transients from a performance stand. It is given as the amount of spare processor capacity at the peak demand, that is during the power spikes that occur on both transients. Outside these peaks THOR maintains real-time performance with more than the required 50% spare capacity. The calculations where performed on a single CPU Pentium 200 MHz processor in the OpenSim NT environment. Available spare capacity will of course increase as faster processors have become available. The core nodalization is indicated in the table; note that the ATHENA T/H model includes the complete NSSS system, i.e. in addition the vessel hydraulics and steam lines; pressurizer and steam generator models are included for PWRs.

Table 1. Performance

<table>
<thead>
<tr>
<th>Transient</th>
<th>System</th>
<th>Nodalization</th>
<th>Minimum Spare Capacity (%)</th>
</tr>
</thead>
<tbody>
<tr>
<td>ATWS</td>
<td>BWR</td>
<td>2220</td>
<td>100</td>
</tr>
<tr>
<td>Rod Ejection</td>
<td>PWR</td>
<td>2316</td>
<td>129</td>
</tr>
</tbody>
</table>
6. SUMMARY

Characteristic features of the neutronics methods employed in THOR are:

- well proven nodal methods that have been in use in engineering design applications over many years
- nuclear data compatibility with standard in-core fuel management code systems
- computationally highly efficient methods that allows for implementation of real-time applications on standard single processor PC computers

Real-time applications within the OpenSim NT training simulator environment were presented and demonstrated the accuracy and performance of THOR.

Because of the general geometry formulation of the neutron kinetics, it is a trivial task to expand the simulator model to a finer mesh for special applications, which will ensure the compatibility of the neutronics model with standard 3-D engineering design codes. Furthermore, with the rapidly improving performance/cost ratio for standard computer equipment, THOR can be easily upgraded to models with higher geometrical resolution of the core in the training simulators.
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COUPLED NEUTRONICS AND THERMAL HYDRAULICS MODELLING IN REACTOR DYNAMICS CODES TRAB-3D AND HEXTRAN

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Abstract

The reactor dynamics codes for transient and accident analyses inherently include the coupling of neutronics and thermal hydraulics modelling. In Finland a number of codes with 1D and 3D neutronic models have been developed, which include models also for the cooling circuits. They have been used mainly for the needs of Finnish power plants, but some of the codes have also been utilized elsewhere. The continuous validation, simultaneous development, and experiences obtained in commercial applications have considerably improved the performance and range of application of the codes. The fast operation of the codes has enabled realistic analysis of 3D core combined to a full model of the cooling circuit even in such long reactivity scenarios as ATWS.

The reactor dynamics methods are developed further and new more detailed models are created for tasks related to increased safety requirements. For thermal hydraulics calculations, an accurate general flow model based on a new solution method has been developed. Although mainly intended for analysis purposes, the reactor dynamics codes also provide reference solutions for simulator applications. As computer capability increases, these more sophisticated methods can be taken into use also in simulator environments.

1. Finnish reactor dynamics codes

In reactor dynamics applications the combination of neutron-physical phenomena with thermal-hydraulics is inherent in the core (Figure 1) and the modelling of the whole cooling circuit is of vital importance. It has therefore been a nearly built-in property of the Finnish reactor analysis codes, or in the combinations of core models with the circuit thermal-hydraulics models, from the first applications to the present day 3D models.
Figure 1. Coupling of physical processes in the core of a light water reactor.

A 3D reactor dynamics code called TRAB-3D (Kaloinnen & Kyrki-Rajamäki 1997) has recently been developed for rectangular lattice light water reactors, and is now in the validation phase. TRAB-3D is based on the one-dimensional BWR dynamics code TRAB (Rajamäki 1980, Rätti et al. 1991), which is continuously used in the safety analyses of the Finnish Olkiluoto power plant. It models the core and the main circulation system inside the reactor vessel, including the steam dome with related systems, steam lines, recirculation pumps, incoming and outgoing flows, as well as control and protection systems. Typical applications are the pump trip and the steam line isolation. The code has also been used for the simulation of the RBMK type reactor in accident conditions (Vanttola & Rajamäki 1989). The OECD/NEACRP dynamic benchmark problems of PWR control rod ejection accidents and control rod withdrawal have been successfully calculated with TRAB-3D. The code is particularly aimed for the analysis of the Olkiluoto BWR plant, but it can also be used for PWRs - the cooling circuits including vertical steam generators can be modelled with a proper circuit model.
HEXTRAN (Kyrki-Rajamäki 1995) is a core dynamics model for hexagonal geometries. It has been widely used as a dynamics analysis tool for the VVER reactors with applications both in the Finnish Loviisa and in the Hungarian Paks VVER-440 safety analyses, and in the analyses of the new Russian VVER-91 concept.

A fast running simulation code called SMABRE (Miettinen 1985) was developed for thermal-hydraulic accidents using point kinetics in the core modelling. Its range of application extends to cover such types of accidents as small break LOCA, primary-secondary leak, steam line break and most parts of a large break LOCA in VVER plants. In addition to safety analysis, this code has been used as the thermal hydraulic model of various NPP full scope and compact simulators. Its models serve as the cooling circuit model in the HEXTRAN code for VVER reactor dynamics analysis. SMABRE models will also be used for circuit modelling of TRAB-3D in PWR applications.

2. Methodology in TRAB-3D and HEXTRAN

In the development of the dynamics codes large emphasis has been put on the numerical methods used, their effectivity and stability.

The 3D neutron kinetics models solve the two-group diffusion equations in homogenized fuel assembly geometry with a sophisticated, very fast two-level nodal method (Kaloinen et al. 1981). The accuracy of the solution method is comparable to the accuracy of advanced nodal methods in static core design codes, but still the codes are fast enough for analyses of longer transients with 3-D neutronics. Besides full core calculations, calculations utilizing core symmetries of half-core, 1/3 and 1/6 in hexagonal and 1/4 in rectangular core geometry can be carried out. Within nodes the two group fluxes are represented by linear combinations of two spatial modes, the fundamental and the transient mode of solution. The dynamic equations include six groups of delayed neutrons. The feedback effects from xenon-poisoning, fuel temperature, moderator density and temperature, and soluble boron density are included in the code. A special treatment has been developed for the dynamic calculation of the moving fuel assemblies of big follower-type control elements which are used in the VVER-440 type reactors.

The reactor dynamics analyses are supported by the reactor physics calculation system of VTT. Reactor physics data for the dynamics codes can be generated starting from the basic nuclear data libraries. The same data is used also in VTT’s simulator applications with APROS (Puska et al. 1997). The flexible description of nodal two-group cross sections in the dynamics codes allows individual dependence of all cross sections polynomially on the feedback effects. The cross sections, their feedback parameters and kinetic parameters are tabulated according to burnup and mean historical density of
moderator, and the nominal cross sections depending on burnup and history variables are calculated and stored in the initial state for every node.

In the reactor core the thermal hydraulics calculation is performed in separate hydraulic channels which can be freely connected with one or more fuel assemblies. The calculation model may include a core bypass channel and unheated channel regions below and above the core. Channel hydraulics is based on the conservation equations for steam and water mass, total enthalpy and total momentum, and on a selection of correlations. The pressure balance over the core determines the distribution of mass flow through the channels, and the phase velocities may be related by an algebraic slip ratio or by the drift flux formalism. Water and steam properties are calculated from rational functions of pressure and enthalpy.

During the hydraulics iterations a one-dimensional heat transfer calculation with several radial mesh points is made for an average fuel rod at different axial elevations in each fuel assembly. The thermal properties of fuel pellet, gas gap and fuel cladding are functions of temperature, and the heat transfer coefficient depends on the hydraulic regime. The fission power is divided into prompt and delayed fractions, and part of the power can be dissipated into heat directly in the coolant.

Advanced time integration methods are used. Time discretization is made by implicit methods which allow flexible choices of time steps. The numerical method can be varied between the standard fully implicit theta method and the central-difference theta method in the fuel and cladding heat transfer and in the thermal hydraulics.

The process description of the VVER circuit model SMABRE in connection with HEXTRAN is based on generalized nodes, junctions connecting nodes and heat structures describing structure walls, fuel rods and steam generator tubes, similarly as in RELAP. The spatial discretization is based on donor cell method and for time discretization a non-iterative semi-implicit algorithm based on predictor-corrector method is used. The five flow equations are solved by applying sparse matrix methods. These methods make the code very fast.

In BWR calculations, both the separate parallel core channels and the cooling circuits are calculated with TRAB-3D using its own thermal hydraulics solution methods. For PWR calculations SMABRE circuit models could be used similarly as in HEXTRAN connection.

The neutronics and thermal hydraulics are strongly coupled in the reactor core and mutual iterations are needed to achieve a stable solution. The solution method of SMABRE is non-iterative and there is a loose coupling between it and HEXTRAN (or TRAB-3D), therefore no iterations are made with the circuit hydraulics.
The coupled code HEXTRAN/SMABRE has its own main program and some connecting subprograms, but as a rule the subprograms of HEXTRAN and SMABRE are used in the same way as in the separate codes. Both codes use their own input, output, restart and plotting capabilities. Thereby the versatility of the codes is not lost and all revisions made in the codes separately can directly be included in the coupled code. The first applications with the coupled code were carried out as early as in 1991 - 1992.

With HEXTRAN and TRAB-3D it is possible to perform fully realistic time-dependent analyses starting from the actual core cycle conditions of the nuclear power plant. The same cross section data can be used as for burnup simulation. Methods for making conservative accident analyses with these best-estimate codes have also been developed. Complicated transients and accidents in which there are strong interactions between neutron kinetics and thermal hydraulics can be reliably analyzed. Due to the effective methods the coupled codes are so fast that even the longest accidents, e.g. ATWS cases, can practically be analyzed with them.

### 3. Applications of coupled 3D neutronics and circuit thermal hydraulics

The rectangular TRAB-3D is a new code, therefore the applications for BWRs have been made so far with the 1D TRAB code, see Table 1. In the following some remarks are given of the applications with 3D HEXTRAN for VVERs. The typical accidents analyzed for VVERs are the same as for the other types of PWRs.

HEXTRAN coupled with SMABRE has been extensively used in contract research both on design basis accidents and on new more exotic areas of, e.g., boron dilution and ATWS accidents, see Table 2. In typical ATWS cases, such as Loss of Feed Water, Loss of Outside Power, or Control Group Withdrawal 1/6 core symmetry can be used in HEXTRAN. However, even whole core geometry has been used in some ATWS cases which allows arbitrary asymmetry in the core, e.g., due to different boron injections to different loops. In more complicated ATWS cases there can be radial asymmetry already in the original disturbance causing the accident (Small Break LOCA, Control Rod Ejection). Typically the calculation model includes about 8000 nodes in the core and about 1000 nodes in the coolant circuits.

A typical example of a design basis accident which cannot easily be calculated without a three-dimensional core model is the control rod ejection accident, where considerable deformation of the radial and axial power distributions occur in the core. An application
Table 1. TRAB applications for BWRs

<table>
<thead>
<tr>
<th>Year</th>
<th>Client and country</th>
<th>Project description</th>
<th>Services provided</th>
</tr>
</thead>
<tbody>
<tr>
<td>1990-1997</td>
<td>Radiation and Nuclear Safety Authority, Finland</td>
<td>Olkiluoto NPP ABB type BWR</td>
<td>Code development and validation, transient analyses (eg RIA, ATWS)</td>
</tr>
<tr>
<td>1982-1997</td>
<td>Teollisuuden Voima Oy, Finland</td>
<td>Olkiluoto NPP ABB type BWR</td>
<td>Licensing transient and accident analyses (eg pump trip, overpressurization)</td>
</tr>
<tr>
<td>1986-1988</td>
<td>Ministry of Trade and Industry, Finland</td>
<td>Accident behaviour of a graphite moderated channel reactor RBMK</td>
<td>Simulation of Chernobyl accident scenarios</td>
</tr>
</tbody>
</table>

Table 2. HEXTRAN nuclear consulting applications

<table>
<thead>
<tr>
<th>Year</th>
<th>Client and country</th>
<th>Project description</th>
<th>Services provided</th>
</tr>
</thead>
<tbody>
<tr>
<td>1989-1997</td>
<td>Radiation and Nuclear Safety Authority, Finland</td>
<td>Loviisa nuclear power plant VVER-440</td>
<td>Code development and validation, transient and accident analyses, ATWS</td>
</tr>
<tr>
<td>1993-1997</td>
<td>Imatran Voima and IVO Power Engineering, Finland</td>
<td>Loviisa nuclear power plant VVER-440</td>
<td>RIA analyses: Boron dilution, rod ejection, steam line break analyses; ATWS</td>
</tr>
<tr>
<td>1992-1996</td>
<td>Atomenergoexport, Russia</td>
<td>VVER-1000/type 91 nuclear power plant concept</td>
<td>Reactivity Initiated Accident analyses, boron dilution, ATWS</td>
</tr>
<tr>
<td>1993-1994</td>
<td>NPP Paks and KFKI, Hungary</td>
<td>Paks nuclear power plant VVER-440, AGNES project</td>
<td>Control rod ejection and steam line break accident analyses</td>
</tr>
</tbody>
</table>
of HEXTRAN for the analysis of control rod ejection accidents is presented by Gács et al. 1995. The coupling between neutronic and thermal hydraulic modelling in the core is especially important in such cases where the coolant mass flow in the initial state is lower than nominal, and there occurs boiling and even DNB in the hottest channels. The main phenomena during a control rod ejection accident (fission power peak, time of trip signals, fuel temperature increase, extent of departure from nucleate boiling occurrence) could be analyzed without modelling the cooling circuits. The accident is so fast that the core mass flow does not markedly change during the critical phase of the accident. However, the simultaneous prediction of the pressure increase can only be made with the full cooling circuit models included in the calculation.

![Graph](image)

*Figure 2. Radial fission power distribution with HEXTRAN during recriticality phase in a steam line break accident. One control rod is assumed to be stuck in the upper position.*

Steam line break is another type of design basis accident where significant radial deformation of fission power occurs in the core. Here the disturbance originates from the cooling circuits and this type of accident cannot be analyzed without modelling both the core and the circuits together. During the possible recriticality phase due to the large overcooling of the core after the trip, the assumption of a control rod stuck in the upper position can further distort the fission power distribution; hot channel factor can be as high as twelve (Figure 2). Although the total power level would not increase to the
Figure 3. Core fission power with HEXTRAN during restart of the first reactor coolant pump in a local inhomogeneous boron dilution scenario in shutdown conditions.

Figure 4. Maximal temperatures in hot fuel rod with HEXTRAN during restart of the first reactor coolant pump in a local inhomogeneous boron dilution scenario in shutdown conditions.
nominal level, the conditions in the hottest part of the core must be carefully analyzed with the three-dimensional dynamics calculation and with additional hot channel calculations, because fuel overheating could occur due to departure from nucleate boiling. A three-dimensional model is also much more reliable in predicting the reactivity level after the trip. HEXTRAN has been applied in steam line break analyses for different plants, where usually the search for most critical cases has first been made with numerous separate SMABRE calculations using point kinetics.

Inhomogeneous boron dilution has recently been found out to be a possible cause of reactivity initiated accidents in PWRs. Extensive studies on its consequences have been made in Finland (e.g. Antila et al. 1991, Siltanen et al. 1997, Gromov et al. 1996). Coupled 3D neutronics - thermal hydraulics code can be utilized principally in two different ways in these studies. Firstly, the consequences of external dilution slugs flowing into the core can be studied. The flow velocity can vary from almost nominal speed to the natural circulation conditions. The initial state of the reactor can vary from almost nominal fission power level to subcritical shutdown conditions. In the coupled code system, most of the effects of the numerical diffusion to the boron dilution front can be eliminated by assuming a dilution straightly in the inlet of the core. In Figures 3 and 4 HEXTRAN results of a local inhomogeneous boron dilution scenario in shutdown conditions are shown during restart of the first reactor coolant pump. Secondly, the coupled code can be used to study such long accidents during which inherent boron dilution in the primary circuit could occur due to boiling/condensation cooling mode (Hyvärinen 1996). Such natural circulation conditions can be achieved e.g. in Small Break LOCA or ATWS cases. In such conditions the formation of local boron dilution slugs cannot be reliably predicted with conventional hydraulics models.

4. Development of new hydraulics models and solution methods

In LWR accident analyses, many complicated phenomena arise which cannot be correctly solved by the hydraulic solution methods applied so far in the world. Presently the thermal hydraulics models of the reactor dynamics codes of VTT Energy are being further developed based on the new PLIM method (Rajamäki & Saarinen 1991, 1994 a). The superiority of the PLIM algorithm in calculating propagation of a boron front through the reactor core in natural circulation conditions has been demonstrated with HEXTRAN (Rajamäki & Kyrki-Rajamäki 1997).

The new physically based two-phase flow model SFAV have been successfully tested against measurements and analytical solution (Rajamäki & Saarinen 1994 b, Narumo & Rajamäki 1995).
4.1 PLIM solution method

PLIM, Piecewise Linear Interpolation Method, is a new highly accurate shape-preserving characteristics method for solving systems of one-dimensional hyperbolic partial differential equations (Rajamäki & Saarinen 1991, 1994a). PLIM is applicable and accurate always when conventional methods are accurate and is able to treat propagating piecewise linear distributions accurately on a mesh grid. In the one-dimensional time-dependent case, interpolation with the piecewise linear polynomial approximation containing two unknown parameters yields the desired shape preserving scheme. The conservation laws are not violated either. The discretization mesh needed and the numerical performance of the solution are in direct proportion to the physical complexity.

PLIM method has been successfully tested in several demanding flow problems, e.g. stratified two phase flow, gas dynamics and various convection diffusion problems (Rajamäki & Saarinen 1994c, Saarinen 1994). The numerical solution can handle all cases of reversed flow. Strong interactions due to source terms of the flow equations are allowed and movable discontinuities such as water levels can appear or disappear.

The new hydraulics solution method has been included in the core channel models of the HEXTRAN and TRAB codes. First results with the HEXTRAN-PLIM code on a challenging three-dimensional boron dilution benchmark (Kyrki-Rajamäki 1996) including boiling have been promising. Most features needed in the circuit models - discontinuities, pumps, connecting flow path subregions into flow paths, nodes connecting flow paths, the closing of the whole circuit and its pressure balance - have been tested.

4.2 SFAV two-phase flow model

A new formalism for two-phase flow has been derived: Separation of the Flow According to Velocity (Rajamäki & Saarinen 1994b, Rajamäki & Narumo 1995, Narumo & Rajamäki 1995, 1996a, 1996b, Narumo 1997a, 1997b). SFAV two-phase flow model consists of six conservation equations for the mass, momentum and energy of two fluids like any conventional two-phase flow model. The difference has been made with the derivation of the momentum equations: the flow is separated into two velocity fields and conservation equations are derived over their domains. The basis is clearly physical: mass and energy are distributed approximately according to the phases, but the variations of momentum follow more the flow velocity.

In the SFAV-formalism the distributions of the phases and their velocities are not treated as uniform in the cross-sectional areas occupied by the fluids, but transverse
spatial dependencies are allowed. With suitable dependencies and without using nonphysical fittings, the equation system is well-posed; in other words, the characteristic velocities are real. The couplings to unknown variables create obstacles to numerical solution, but they are easily overcome with the new shape-preserving solution method PLIM. The SFAV model has been shown to give sonic velocities in very good agreement with measurements over a full range of void fractions. The calculated propagating velocities of void fraction disturbances agree well with measurements.

The characteristic velocities of the SFAV-equation system can be associated with the propagation velocities of small disturbances. They have been shown to be real in any flow conditions. This does not prevent a model to predict physical instabilities in case needed. In addition, the characteristics have been verified to be very well in agreement with measurement data obtained at low pressures; relevant data at high pressure has not been available yet.

As an application, response of the flow in a partial BWR fuel channel has been calculated as function of frequency of a sine-wave inlet disturbance (Narumo 1996a). The results have been compared to those obtained with the corresponding drift-flux model. This kind of situation could happen in practise e.g. with feedwater pump partial damage or blockage in the fuel channel. The results obtained by each model differed from each other already with circa 5 Hz frequency, while the time constant for the relaxation of steady-state distributions would have implied that significant differences do not appear until with frequencies of order 50 Hz. This may be an evidence that accurate modelling of flow dynamics is necessary also for fairly slow transients. In the future SFAV model will be applied in the reactor dynamics codes of VTT which have now four or five equations in their hydraulics model. Especially in the calculation of film boiling in hot channel analyses, the six equation model would be preferable (Vanttola 1993).

5. Conclusions

In VTT reactor dynamics codes with 3D neutronic models have been developed, which include models also for the cooling circuits. Demanding scenarios for such complex transients as ATWS can be studied with reasonable computing effort, using an advanced three-dimensional core model and a detailed circuit model, with accuracy sufficient for safety analyses. At present most of the licensing analyses of the Finnish NPP:s can be performed with our own codes, which have also been applied in the safety analysis of some foreign plants. The reactor dynamics methods are developed and new more detailed models are created for tasks related to increased safety requirements. Although mainly intended for analysis purposes, the reactor dynamics codes also provide reference solutions for simulator applications. As computer capability increases, these more sophisticated methods can be taken into use also in simulator environments.
References


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APROS 3-D CORE MODELS FOR SIMULATORS AND PLANT ANALYZERS

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Abstract

The 3-D core models of APROS simulation environment can be used in simulator and plant analyzer applications, as well as in safety analysis. The key feature of APROS models is that the same physical models can be used in all applications. For three-dimensional reactor cores the APROS models cover both quadratic BWR and PWR cores and the hexagonal lattice VVER-type cores. In APROS environment the user can select the number of flow channels in the core and either five- or six-equation thermal hydraulic model for these channels. The thermal hydraulic model and the channel description have a decisive effect on the calculation time of the 3-D core model and thus just these selections make at present the major difference between a safety analysis model and a training simulator model. The paper presents examples of various types of 3-D LWR-type core descriptions for simulator and plant analyzer use and discusses the differences of calculation speed and physical results between a typical safety analysis model description and a real-time simulator model description in transients.

1. Introduction

Three-dimensional APROS core models have traditionally been used in engineering simulators and in plant safety analysis. Due to the increase of computer performance creation of three-dimensional core models with real time calculation capability is already possible, which extends the use of APROS three-dimensional core models also for training simulator applications.

APROS software consists of physical models that are grouped into general and application specific packages (Juslin and Paljakka 1997). The software is designed to be used with graphical user interface. A typical feature of the software is the requirement
for on-line modifications in physical parameters and also in model structure during the simulation. These properties have been built in the three-dimensional core models as well.

In APROS the essential choices between the various three-dimensional core applications are the selection of core thermal hydraulic model and division of core into thermal hydraulic channels. The paper describes the main features of the core models in training simulator and plant analyser use versus safety analysis models. The structure of APROS 3-D core models and the coupling of the thermal hydraulics and neutronics in core is described, too. The effects of the two alternative thermal hydraulic models and various core divisions on calculation time and physical results for the various core types is also discussed. The example calculations have been performed with APROS version 4.05 using a HP-180 C workstation.

2. Core models in simulators, plant analyzers and safety analysis

Figure 1 indicates various possibilities to use APROS simulation environment in nuclear applications starting from the design of a new plant and ending in the use of APROS during the plant lifetime for training, planning of plant design changes and update of plant safety analysis (Tiihonen and Justil 1996).

![Diagram](image)

*Figure 1. Use of APROS for engineering and training simulator applications and safety analysis during plant lifetime.*
In APROS the 3-D core neutronics is always described with a finite difference type model and the thermal hydraulics with the five- or six-equation model. For a training simulator application the required real-time or faster than real time calculation speed is obtained with lumping large parts of the core together both in neutronics and in thermal hydraulics and with selection of the faster thermal hydraulic model, the five-equation model, for the thermal hydraulic channels. In training simulator applications typically 10-15 axial nodes are used both in neutronics and in thermal hydraulics. In order to gain speed the identical 4-8 fuel assemblies of various core types are usually placed in the same thermal hydraulic channel. Especially in BWR applications with a large number of fuel assemblies the fuel assemblies in radial direction can further be lumped into so-called macroelements. The asymmetric effects in core can be presented with a small group of fuel assemblies placed in separate flow channels (the concerned assembly and the closest 4-6 neighbours). Further, the connection of the core thermal hydraulic channels with the parts below and above the core region must be described in a realistic manner. The calculation speed of quite detailed plant process and automation descriptions with APROS with one-dimensional core model have reached faster than real time performance already some years ago (Puska and Kontio 1994a). A further way to ensure real time performance with the three-dimensional core is to use the parallel APROS application with the plant process and automation on one workstation and the 3-D core model on another workstation. The second requirement of a training simulator: the fidelity to plant response, is well ensured by the detailed physical models in core and possibility for detailed plant-like description with the process and automation components.

An engineering simulator or plant analyzer is used for instance when planning replacement of some part of an existing power plant. In this application the detailed and true description of those parts concerned in the change is essential. For operations affecting the core this requires the description of the core with the level of detailisation available in core input data axially according to the available burnup and enrichment distribution and describing each fuel assembly separately in radial direction. In axial direction the number of nodes extends usually from 10 to 30. For the description of asymmetric transients each fuel assembly must be placed in a separate thermal hydraulic channel. Also in the plant ana lys er the proper connection of the core to the process and automation system is essential. In a plant analyser, too, the choice of five-equation thermal hydraulic model and application of parallel core and process calculation on two workstations increases the calculation speed.

In safety analysis the most essential feature is the reliable result that must be calculated with taking into account all the required conservativity features in the analysis. Thus for the description of asymmetric core transients originating from events in the core, like control rod ejection, or outside the core, like erroneous start of a main circulation pump, the detailed description of core with fuel assemblies placed in separate flow channels is
essential. Safety analysis may also require use of the more detailed six-equation thermal hydraulic model in extreme situations. A typical feature of a safety analysis model is that only those parts of the plant process and automation having an active role in the transient are described.

3. Structure of APROS 3-D core models

3.1 General structure

In APROS the user operates with graphical user interface using components at the process component level. In the three-dimensional core model the process components are the fuel assemblies, the control assemblies, the reflector assemblies and the one-dimensional thermal hydraulic flow channels. The user specifies the connections of the process component into other process components, too. On the basis of the information given at the process component level, the system then creates the calculational level of nodes and branches. During the simulation process the user can change the process component data or change the connections of the process component to other process components without the need of recompilation of the code.

In addition, the 3-D model requires some general information like indication of reactor type (BWR, PWR or VVER) and assembly geometry (hexagonal or quadratic) plus cross section data file.

3.2 Process components

In the three-dimensional core model the process components are the fuel assemblies, the control assemblies, the reflector assemblies and the one-dimensional thermal hydraulic flow channels. For each fuel assembly the user must specify a name, which usually corresponds to some indication code at the actual core, a x-y position in the core model, division of assembly in axial sections, and burn-up, enrichment, and for BWR also void and control rod history in each axial section. Similar information plus indication of rod position is given for each control assembly. In a similar manner, reflector assemblies can be defined, or extrapolation distances can be used instead. Additionally, for each fuel assembly and control assembly the name of the thermal hydraulic channel where it belongs must be specified. In the similar manner for each thermal hydraulic channel a name must be specified along with the type of thermal hydraulic model (five- or six-equation model), fuel rod geometry data, number of fuel rods, flow area of channel, hydraulic diameter, nominal power and division of the channel in the axial direction etc.
A specific feature of APROS is that user can place the assemblies into thermal hydraulic channels according to the requirements of the calculation.

In addition, the 3-D model requires some general information like indication of reactor type (BWR, PWR or VVER) and assembly geometry (hexagonal or quadratic) plus cross section data file. On the basis of the information given the neutronics and thermal hydraulics calculation level descriptions are created by APROS. This includes also creation of the index tables telling which neutronics node is connected to which thermal hydraulic node in the core. All fuel and control assemblies are assumed to have the same size and axial division. These correspond usually to the real core fuel and control assemblies or in some cases to the so-called macro elements. The axial division is usually also quite well defined by the nature of the core input data.

3.3 Physical models

APROS code has one-dimensional and three-dimensional core neutronics models. Both models have two energy groups and six delayed neutron groups. In the models the basic equations are first discretized. In the three-dimensional model the neutron flux equations are integrated over the node volumes, a few approximations are made and the fast and thermal equations are solved using Gauss-Seidel iteration process. The finite-difference type three-dimensional neutronics model is able to describe both hexagonal and quadrilateral fuel assembly geometry (Puska and Kontio 1995). The concentrations of six delayed neutron groups are calculated. Iodine, xenon, samarium and promethium calculations are included, too, with user-selected speedup factor. Reactivity feedback effects due to fuel temperature, coolant density and temperature, coolant void fraction, coolant boron content and control and scram rods are taken into account in the models.

For the one- and three-dimensional core models the user can select the five- or the six-equation thermal hydraulic model (Al-Falahi et al 1995). The five-equation model is based on the conservation equations of mass and energy for liquid and gas phases and momentum equation for mixture of gas and liquid. In the five-equation model the gas and liquid interface friction is not calculated, but the differential phase velocities are obtained through the drift flux correlations. A separate drift flux model calculates the mass flow rates of the phases. The quantities to be solved in the model are pressure, volumetric flows, void fractions and phasial enthalpies. No iteration is needed in the model.

The six-equation model describes the behaviour of one-dimensional two-phase flow. The model is based in the conservation equations of mass, momentum and energy for the gas and liquid phases. The equations are coupled with empirical correlations describing various two-phase phenomena. The pressures and velocities, volume fractions and
enthalpies of each phase are solved from the discretized equations using an iterative procedure. Special correlations are provided for re-flooding phenomena. A moving mesh model for axial heat conduction can be used, if the heat flows have to be calculated with great accuracy.

Heat transfer modules connect the thermal hydraulic models with their own heat conduction solutions. Each thermal hydraulic model contains a boron concentration solution, too. The thermal hydraulic part of APROS contains also calculation of fuel enthalpy and oxide layer thickness on cladding surface and power production by cladding oxidation according to Baker-Just model that are required for the hot channel calculations. The information from the core flow channels to the hot channels is transmitted with the boundary condition modules of thermal hydraulics.

Calculation of fuel rod temperatures is performed in the thermal hydraulic part of APROS. One dimensional heat conduction solution in fuel rod is usually calculated using ten radial nodes. Specific materials properties can be given for the fuel pellet, gap and cladding without the need of re-compiling the code. A temperature dependent gap conductance calculation can be performed too. At present, axial heat conduction in the fuel rod can also be taken into account.

4. Extent and performance of 3-D core models

Table 1 contains typical VVER core descriptions used in APROS for training simulator, plant analyser and safety analysis purposes. Table 2 gives corresponding information for some PWR and BWR core models. The calculation times of the various three-dimensional APROS core models with the same slightly optimised program version using HP-180 C computer have been given in Tables 3 and 4.

The VVER1-VVER4 in Table 1 is VVER-440 type reactor core of Finnish Loviisa plant of the IVO utility. The core has 313 fuel assemblies and 37 control assemblies. In the APROS model the neutronics part is described with 313 hexagonal fuel assemblies that have been divided into ten axial segments. In the models VVER1 and VVER2 of Table 1 six similar fuel assemblies were placed in the same thermal hydraulic channel. The central assembly was placed into its own channel. Thus this alternative contained 53 channels that were described either using the five-equation or the six-equation thermal hydraulic model. These alternatives represent typical core sizes for training simulator use. The most simple way to represent asymmetric effects in training simulator application in APROS is to describe the affected rod and the nearest neighbours with separate thermal hydraulic channels, which for the Loviisa core would mean 7 additional channels thus increasing the total number of thermal hydraulic channels from 53 to 60. In
the models VVER3 and VVER4 each fuel assembly was placed in its own thermal hydraulic channel that was described either with the five- or the six-equation thermal hydraulic model. These core descriptions are suitable for engineering simulator and plant safety analysis.

The VVER5 in Table 1 is a VVER-1000-type core of the advanced VVER-91 design. The core contains 163 hexagonal fuel assemblies and 97 control rod clusters. In the APROS model the core neutronics part has been described with 163 hexagonal fuel assemblies that have been divided into ten axial segments. The model was created for safety analysis (Puska et al. 1997b) and thus one fuel assembly per thermal hydraulic channel was used. Each thermal hydraulic channel was divided into ten sections and described with the five-equation thermal hydraulic model.

Table 1. Three-dimensional APROS VVER-type core models with various thermal hydraulic models and channel divisions.

<table>
<thead>
<tr>
<th>Core type</th>
<th>VVER1</th>
<th>VVER2</th>
<th>VVER3</th>
<th>VVER4</th>
<th>VVER5</th>
</tr>
</thead>
<tbody>
<tr>
<td>Fuel assemblies</td>
<td>313</td>
<td>313</td>
<td>313</td>
<td>313</td>
<td>163</td>
</tr>
<tr>
<td>Axial division</td>
<td>10</td>
<td>10</td>
<td>10</td>
<td>10</td>
<td>10</td>
</tr>
<tr>
<td>5-eq. Channels</td>
<td>53</td>
<td>-</td>
<td>313</td>
<td>-</td>
<td>163</td>
</tr>
<tr>
<td>6-eq. Channels</td>
<td>-</td>
<td>53</td>
<td>-</td>
<td>313</td>
<td></td>
</tr>
<tr>
<td>Neutronics nodes</td>
<td>3130</td>
<td>3130</td>
<td>3130</td>
<td>3130</td>
<td>1630</td>
</tr>
<tr>
<td>Thermal hydraulic nodes</td>
<td>530</td>
<td>530</td>
<td>3130</td>
<td>3130</td>
<td>1630</td>
</tr>
</tbody>
</table>

The BWR1 and BWR2 in Table 2 describe BWR-type core of the Finnish Olkiluoto plant of the TVO utility. The core contains 500 fuel assemblies and 121 control assemblies. In APROS model the neutronics part was described with 500 quadratic fuel assemblies. Two alternatives were used in the description of thermal hydraulic flow channels: either two identical fuel assemblies were placed into each one-dimensional thermal hydraulic channel or four assemblies were placed into same thermal hydraulic channels. Thus the total number of thermal hydraulic channels in the core was 250 or 125, respectively. Axially each fuel assembly and thermal hydraulic channel was divided into 25 sections. In the BWR models the thermal hydraulic channels were described using the five-equation thermal hydraulic model. These core models represent those suitable for safety analysis.

The PWR1 and PWR2 in Table 2 describe PWR-type core used for the calculation of the OECD NEA LWR core transient benchmark (Leppänen 1997). The core has 157 fuel assemblies that were divided into 16 axial segments. The model was created for the calculation of the benchmark, and in the model each fuel assembly was placed into own thermal hydraulic channel. Both the five-equation and the six-equation models were used in the calculation of the transient.
Table 2. Three-dimensional APROS LWR-type core models with various thermal hydraulic models and channel divisions.

<table>
<thead>
<tr>
<th>Core type</th>
<th>BWR1</th>
<th>BWR2</th>
<th>PWR1</th>
<th>PWR2</th>
</tr>
</thead>
<tbody>
<tr>
<td>Fuel assemblies</td>
<td>500</td>
<td>500</td>
<td>157</td>
<td>157</td>
</tr>
<tr>
<td>Axial division</td>
<td>25</td>
<td>25</td>
<td>16</td>
<td>16</td>
</tr>
<tr>
<td>5-eq Channels</td>
<td>125</td>
<td>250</td>
<td>157</td>
<td>-</td>
</tr>
<tr>
<td>6-eq Channels</td>
<td>-</td>
<td>-</td>
<td>-</td>
<td>157</td>
</tr>
<tr>
<td>Neutronics nodes</td>
<td>12500</td>
<td>12500</td>
<td>2512</td>
<td>2512</td>
</tr>
<tr>
<td>Thermal hydraulic nodes</td>
<td>3125</td>
<td>6250</td>
<td>2512</td>
<td>2512</td>
</tr>
</tbody>
</table>

The calculation speed reached with the core descriptions of Table 1 have been presented in Table 3. Time step size 0.2 seconds is usual for training simulators at steady state. In safety analysis shorter time steps are used. It can be observed that for VVER-type reactors real-time calculation speed can at present be reached with cores where the symmetric fuel assemblies have been placed in the same five-equation thermal hydraulic channel. With cores consisting of relatively small number of fuel assemblies, like the VVER-91, even the full core description with one fuel assembly per flow channel is quite fast.

Table 3. Calculation speeds with various VVER cores at steady state with various thermal hydraulic models. Time-step 0.2 s. Computer HP-180C. Speed indicated is simulation time/CPU time.

<table>
<thead>
<tr>
<th>Core type</th>
<th>VVER1</th>
<th>VVER2</th>
<th>VVER3</th>
<th>VVER4</th>
<th>VVER5</th>
</tr>
</thead>
<tbody>
<tr>
<td>Relative calculation speed</td>
<td>1.24</td>
<td>0.78</td>
<td>0.32</td>
<td>0.18</td>
<td>0.66</td>
</tr>
</tbody>
</table>

Table 4 shows the corresponding calculation times obtained with the core descriptions of Table 2. It can be noticed that the calculation speed obtained for the western PWR-core with the 5-equation thermal hydraulic model is quite fast. For the six-equation model the calculation time required is approximately double, as indicated also for the VVER-type cores. For the quite detailed BWR-type cores reaching of real-time calculation would still require reduction of axial division and lumping of fuel assemblies into macroelements and further placing identical macroelements into same thermal hydraulic channel.

Table 4. Calculation speeds with various BWR and PWR cores at steady with various thermal hydraulic models. Time-step 0.2 s. Computer HP-180C. Speed indicated is simulation time/CPU time.

<table>
<thead>
<tr>
<th>Core type</th>
<th>BWR1</th>
<th>BWR2</th>
<th>PWR1</th>
<th>PWR2</th>
</tr>
</thead>
<tbody>
<tr>
<td>Relative calculation speed</td>
<td>0.22</td>
<td>0.14</td>
<td>0.43</td>
<td>0.23</td>
</tr>
</tbody>
</table>
5. APROS 3-D core model applications

The use of three-dimensional core models is self-evident in the area of safety analysis, especially in asymmetric events in the core. APROS 3-D core models have been applied for the safety analysis of VVER-91 type reactors for the calculation of various ATWS cases covering main steam line break, main steam header break and erroneous connection of a main reactor coolant pump (Puska et al. 1997b).

Full three-dimensional core models have been created also for the Finnish Loviisa VVER-440 plant (Puska et al 1995). The power distributions of these core models have been compared with power distributions obtained with other codes at steady state (Puska and Kontio 1994b) and the core models have been used for demonstrating the capability of the APROS models for calculation of control rod and boron dilution type transients.

Recently 3-D APROS core models have been created for the Finnish Olkiluoto BWR plant for transient and severe accident recriticality analyses (Puska et al 1997a). As an example of type of information obtained from three-dimensional core model calculation and the visualization of the results the steady state fast flux, fuel temperature and void fraction results at three core elevations have been presented in Figures 2-4. For this core description a drop of two control assemblies was calculated. The transient resulted into a power peak of over four times the original full power (Puska et al 1997a).

![Figure 2](image)

*Figure 2. Fast flux distribution at core lower part, position of steady state maximum power and at core upper part for a BWR core at steady state.*
Figure 3. Fuel temperature distribution at core lower part, position of steady state maximum power and at core upper part for a BWR core at steady state.

Figure 4. Void fraction distribution at core lower part, position of steady state maximum power and at core upper part for a BWR core at steady state.
Figure 5 shows the fast flux at the time of power peak for the same core levels as in Figure 2. With more advanced visualization means all the information available in each calculation node of the core model can be exposed in a more illustrative way both to the analyst and to the operator.

![Figure 5. Fast flux distribution at core lower part, position of steady state maximum power and at core upper part for a BWR core at time of peak average power.](image)

6. Discussion

There are various means to represent the three-dimensional core with the APROS model for training simulators, plant analyzers and safety analysis. The basic differences are the selection of the thermal hydraulic model for the core and the division of the core into thermal hydraulic channels.

Calculation of the PWR core transients for the NEA LWR benchmark has indicated that there are no practical differences between the results of the five and six-equation thermal hydraulic models in such transients (Leppänen 1997). Similar results have been obtained for the calculations of VVER-440 transients (Puska et al 1994c). For the BWR only five-equation thermal hydraulics has thus far been used. However, the experience obtained from calculating BWR transients with one-dimensional core model and the alternative thermal hydraulic models indicated good agreement in some cases whereas for one severe transient there was a significant difference in the results obtained (Puska et al 1997a). Thus for the training simulator applications the faster five-equation thermal
hydraulic model is the natural choice for PWR and VVER cores. Also for the BWR type training simulators the five-equation thermal hydraulic model is preferable due to the calculation speed. The dynamic 3-D codes HEXTRAN and TRAB with more sophisticated neutronics (Räty and Kyrki-Rajamäki 1997) offer a further possibility to check the physical results obtained with the various APROS 3-D core model alternatives.

Studies performed for various BWR and VVER type cores (Puska and Ylijoki 1997) have indicated that the thermal hydraulics required 60-90% of the calculation time, and the neutronics solution required respectively only 10-40% of the calculation time. The calculations performed with PWR and VVER-type cores indicated that the five-equation thermal hydraulic model was approximately twice faster than the six-equation thermal hydraulic model per node, as observed also in previous studies. The difference in calculation speed originated mainly from the fact that with the five-equation model no iteration was performed whereas with the six equation model at least one iteration round was included.

The calculation time studies indicated that at present the symmetric core descriptions with the five-equation thermal hydraulic channels can already reach real-time performance in steady state calculation with modern workstations. This, in connection of the continuing development of computer technology, indicates that three-dimensional core descriptions are a realistic alternative also in training simulators. Also the 3-D core descriptions with one assembly per five-equation thermal hydraulic channel will soon become a realistic alternative for training and engineering simulators, whereas the core descriptions using the six-equation thermal hydraulic model are expected to find their major application area in the safety and severe accident studies.

7. Concluding remarks

The alternative ways of coupling full 3-D neutronics and thermal hydraulics in APROS enable creation of various 3-D core models for various simulation purposes. The essential choices are the number of flow channels in the core and either five- or six-equation thermal hydraulic model for these channels.

The calculation time studies indicated that at present the symmetric core descriptions with the five-equation thermal hydraulic channels can already reach real-time performance in steady state calculation with modern workstations. This, in connection of the continuing development of computer technology, indicates that three-dimensional core descriptions are a realistic alternative also in training simulators.
In previous work the physical results obtained with the two alternative thermal hydraulic models have been found to be very similar for most transients, and thus the faster five-equation thermal hydraulic model is the natural choice for 3-D training simulator cores. Core descriptions using the six-equation thermal hydraulic model are still expected to find their major application area in the safety and severe accident studies.

References


Effective Modeling of Hydrogen Mixing and Catalytic Recombination in Containment Atmosphere with an Eulerian Containment Code

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Abstract

Large amounts of hydrogen can be generated in the containment of a nuclear power plant following a postulated accident with significant fuel damage.

Different strategies have been proposed and implemented to prevent violent hydrogen combustion. An attractive one aims to eliminate hydrogen without burning processes; it is based on the use of catalytic hydrogen recombiners.

This paper describes a simulation methodology which is being developed by Ansaldo, to support the application of the above strategy, in the frame of two projects sponsored by the Commission of the European Communities within the IV Framework Program on Reactor Safety. Involved organizations also include the DCMN of Pisa University (Italy), Battelle Institute and GRS (Germany), Politecnical University of Madrid (Spain).

The aims to make available a simulation approach, suitable for use for containment design at industrial level (i.e. with reasonable computer running time) and capable to correctly capture the relevant phenomenologies (e.g. multiflow convective flow patterns, hydrogen, air and steam distribution in the containment atmosphere as determined by containment structures and geometries as well as by heat and mass sources and sinks).

Eulerian algorithms provide the capability of three dimensional modelling with a fairly accurate prediction, however lower than CFD codes with a full Navier Stokes formulation. Open linking of an Eulerian code as GOTHIC to a full Navier Stokes CFD code as CFX 4.1 allows to dinamically tune the solving strategies of the Eulerian code itself.

The effort in progress is an application of this innovative methodology to detailed hydrogen recombination simulation and a validation of the approach itself by reproducing experimental data.
1 Introduction

As far as computer codes for containment thermalhydraulic analysis are concerned, the following three levels of approach can be in principle adopted, with the associated pros and cons:

- Traditional lumped parameters models require a limited computer time and have shown an acceptable quality of predictions for a variety of containment simulations. However, problems exist in modelling local phenomena, non homogeneous conditions and momentum transport. In particular there is a serious limitation in applying such codes to a multifluid interaction (buoyancy). In fact it is generally acknowledged that containment analysis codes based on lumped parameters models tend to underestimate gradients in light gases concentrations (i.e. they tend to predict rather homogeneous concentrations).

- Distributed parameters codes, based on the Eulerian approximation of the momentum equation, even if do not have the capability for multifluid treatment, tend to significantly increase the accuracy of the predictions. Such behaviour has been confirmed, for example, by the application of the GOTHIC code to HDR experiments (Wolf/1/, Fischer/2/). Moreover, the computer time is still quite acceptable.

- Presently available CFD codes provide, in principle, a much higher capability since the full NAVIER STOKES formulation allows detailed representation of momentum transport for a multifluid mixture. However they do not have the capability for multiphase treatment. The heat exchange with the structures, under condensing conditions, is oversimplified as steam sink term, there is no conservation equation for the condensate and therefore evaporation cannot be modelled in a meaningful way. Moreover their computational inefficiency discourages their use for transient analysis, apart from specific applications with relatively simple meshes.

Based on the above, the adoption of distributed parameters models thermalhydraulic codes, based on the Eulerian formulation of the momentum equation, is believed to be the most balanced approach for the design and verification of containment system mitigation measures.

The paper describes the steps already performed to extend the capability of the chosen Eulerian code (GOTHIC) to simulate the performance of catalytic hydrogen recombiners and their effect on the convective motions in a given, progressively more complex, geometry.

Such steps include:

- empirical recombiner performance model
- recombiner thermohydraulic model
- prediction of convective motions in progressively more complex geometries
• simulation of multirecombiner experiments in the multicompartment Battelle Model Containment.

They are part of a more general effort, the final aim of which is to develop and assess a methodology, suitable for a generic multidimensional multipurpose Eulerian computer code, capable to effectively predict hydrogen distribution in a real plant containment.

The basic idea is to use a full Navier Stokes CFD code, run with the steady state option at a given time, as reference code for a specific containment geometry and phenomenological situation, including multifluid momentum transport. One effective way to use it, for a containment analysis, is just by fixing heat transfer as well as evaporation and condensation rates as boundary conditions. The usefulness of such approach derives from the great generality of a CFD code, the accuracy of which should be practically independent from the geometry, once the proper noding mesh and code basic options, adequate to capture the phenomenologies of interest, are determined and validated.

The needed boundary conditions, at the given time, should be provided by one of today available and suitable containment Eulerian thermalhydraulic codes (Gothic, Raloc Bassim, Melcor, etc etc.).

The basic assumption is that time histories of hydrogen mixing and distribution, as well as of the circulation flow paths in the containment atmosphere, are relatively slow transients for which the time derivatives are negligible (quasi steady state). This is certainly true at almost any time instant along the evolution of the conceivable accident sequences. This would allow to “freeze”, at the given time, the simulation performed by the Eulerian code, run the CFD code as steady state with the proper boundary conditions and then compare the flow patterns and gases distribution in the containment atmosphere at that given time.

2 Hydrogen Recombiner Model in the Eulerian Code

2.1 Empirical recombiner performance model

The recombiner has been implemented in the GOTHIC code as a new code component; its performance is based on the empirical correlations, developed by the DCMN of Pisa University, between the hydrogen recombination rate and H₂ molar density, as shown in fig.1 for two recombiners of the Zx series of experiments performed in the Battelle Model Containment (/3/) and used to validate the methodology.

The recombiner performance is described by an exponential law:

\[ R_e = A \cdot B^C \]
where:
\[ R_r = \text{recombination rate (recombined hydrogen mass versus time, Kg/h)} \]
\[ A, C = \text{constant coefficients, depending on the recombiner component} \]
\[ B = \text{hydrogen molar concentration (mol/m}^3\). \]

Mass and energy conservation implies also the simulation of the effects of the chemical reaction between oxygen and hydrogen.

### 2.2 Recombiner thermohydraulic model

The strategy in developing the recombiner model for GOTHIC has been focused to create a recombiner component model which allows the heat source to be located outside the subdivided volumes in order to gain as much flexibility as possible to model complex calculation domains where lumped and distributed parameters can be used jointly.

The choice has been that the hydrogen recombiner component is located always on a junction not connected to a boundary condition and that this component converts a specific fraction of the hydrogen (and oxygen) flowing through the junction to steam. The junction is simply a flow connection between any two cells of the same subdivided volume.

The GOTHIC mass, energy and momentum conservation equations have been modified to take into account elimination of \( \text{H}_2 \) and \( \text{O}_2 \) as well as generation of \( \text{H}_2\text{O} \) and heat to properly simulate the onset of buoyancy driven motions due to the recombiner operation.

It has been judged realistic the assumption that the liquid and drop content in the volume around the recombiner is not altered by the catalytic reaction but, due to the high temperature inside the recombiner, only vapor exits from the recombiner once the recombination reaction starts.

Concerning the fluid energy equations, the vapor energy balance has been modified in order to take into account the generation of thermal energy from the recombination reaction.

Then the momentum equation for the junction has been modified in order to include as momentum source also the buoyancy driven force due to the recombiner operation, since the recombiner is seen by the code as a piece of equipment that creates a temperature increment between inlet and outlet.

The simulation of the relevant recombiner geometrical features has also been set-up. To the purpose, since the recombiner model consists of two junctions (the first simulating the recombiner inlet, the second simulating the recombiner cartridge and outlet) connected to a lumped volume with a free volume equal to the free volume of the recombiner, in order to
properly evaluate the pressure drop through the junction where the recombinder is operating, the following assumptions have been made:

1) The distributed pressure drops are essentially due to the laminar flow inside the recombinder cartridge.

2) The concentrated pressure drops are essentially due to the exiting flow from the recombinder since the gas velocity increases when crossing the recombinder cartridge.

3) The recombinder cannot work in reverse flow conditions.

4) If drop and/or liquid phases are present into the junction when the recombinder starts to operate they flow out at the same velocity of the vapor.

5) The average temperature inside the cartridge is calculated as:

\[ T_0 + 0.5 \Delta T \]

where \( T_0 \) is the gas temperature at the recombinder cartridge inlet and \( \Delta T \) the overall temperature increase across the recombinder.

6) The gas viscosity and density, evaluated by using the gas average temperature inside the cartridge, are used for the evaluation of the pressure drop across the cartridge.

7) The distributed pressure drop coefficient inside the cartridge is evaluated by choosing, at each time step, the maximum between these two values:

- 64/Re (laminar flow)
- 0.0055 + 0.55*Re^(-0.333) (turbulent flow)

(at this time roughness or form factor of the cartridge are not accounted for).

8) The recombinder outlet temperature is the recombinder inlet temperature plus \( \Delta T \).

9) The gas density, evaluated by using the cartridge outlet temperature, is used for the evaluation of the concentrated pressure drop.

10) The concentrated pressure drop coefficient only represents the geometrical restriction or expansion that the hot mixture, exiting the recombinder cartridge, encounters.

11) Concentrated pressure losses due to the flow entering both the recombinder and the recombinder cartridge are simulated with a separate junction.

A significant effort has been required to properly address the pressure loss calculation across the recombinder, due to the peculiar flow field inside the recombinder cartridge, characterized by a low Reynolds number which, in addition, significantly changes during recombinder operation.

The mixture (gas plus steam) flowing inside the recombinder when crossing the recombinder cartridge increases its velocity and its temperature due to the recombination reaction heat.

The mixture temperature increment causes a consequent increment of its viscosity with a strong feedback on the Reynolds number and on the type of flow inside the recombinder. Then the pressure drop coefficient is a function of the flow field itself. Hence, in order to properly
simulate the behaviour of the recombinner a refined model for the evaluation of the pressure drops is required to be implemented into the GOTHIC solver.

3 Hydrogen Mixing and Recombination in a Single Volume

3.1 Convective motions in a single volume

A model volume of 630 m$^3$, shaped like a parallelepiped and containing a catalytic hydrogen recombinder (manufactured by NIS), has been initially described by means of a GOTHIC 3D model to perform sensitivity analyses on the convective flow patterns. The initial aim was to identify a nodalization capable to correctly capture convective motions around a heat source. The GOTHIC model is based on an (XxYxZ) grid, that is it includes (XxY) parallel vertical channels with Z vertical subdivisions. The sensitivities showed that although the scheme of the predicted convective motions did not qualitatively change when moving from the initial (5x5x5) grid to a (7x7x7) or to a (3x3x6) grid, the required computer time showed an increase of a factor of about 10 or a decrease of a factor of about 3 respectively. Thus, having in mind that the ultimate aim of this work is to apply the simulation methodology to a real containment volume with a large number of catalytic recombinder, it was decided to adopt the (3x3x6) grid.

With this grid additional sensitivity analyses were performed with the aim to confirm the adequacy of the recombinder model.

The following physical parameters were varied:

- hydrogen initial concentration in the volume (2%, 4%, 8%)
- recombinder chimney height (0.0 m, 0.26 m, 0.76 m)
- recombinder concentrated loss coefficients (50%, 100%, 200% of reference value)
- recombinder geometrical location in the volume (1 m, 3 m, 12 m, above volume floor).

Each simulation covered a time interval of one hour during which no hydrogen was injected into the volume. Hence the hydrogen concentration was constantly decreasing from the initial value (the volume temperature was allowed to slightly increase at the beginning of the transient and then was controlled at an almost constant value).

In the absence of specific experimental data for comparison of code predictions, an engineering judgment was used to determine the consistency of the following calculated parameters:

- recombinder inlet flow velocity
- recombinder inlet and outlet fluid temperature
- recombinder efficiency
- recombinder inlet hydrogen mass flowrate
- recombinder hydrogen mass flowrate

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hydrogen distribution in the volume.

Some of the above parameters, as calculated by the modified GOTHIC when varying the recombining location in the volume from 1 m above (i.e. very close to) the floor to 12 m above the floor (i.e. at about half-way to the ceiling) are summarized in Figures 2 to 7 and commented below.

Fig. 2 shows the recombining inlet flow velocities when the recombining device locations are, 1m (base case), 3m and 12m from the volume bottom respectively. The figure shows that the inlet flow velocity does not significantly depend from the device location at the beginning of the transient; a lower inlet velocity is noted later in the transient when the device is located at 12m from the bottom. This is due to the hydrogen stratification predicted below the device that reduces the hydrogen concentration above it and hence the hydrogen mass passing through the recombining itself more quickly during the transient (fig. 6 and 7).

The inlet flow velocities become very similar again at the end of the transient when most of the hydrogen above the recombining elevation has been recombined.

Fig. 3 shows the temperature of the fluid entering and exiting the recombining (this one calculated immediately after the catalyst cartridges) when the recombining device locations is 1m from the bottom. The temperature increase across the recombining is consistent with the predicted inlet flowrate and hydrogen recombination rate (fig. 5).

In Fig. 4 the comparison between the efficiencies related to the different recombining locations, is reported. The results show that the calculated efficiency does not directly depend from the device location.

Fig. 5 shows, for a recombining located 1m above the floor, the following parameters:
- effective (driven by convective force) hydrogen mass flowrate
- hydrogen recombination rate as calculated based on the empirical correlation developed by DCMN.

During all the transient the predicted convective motions ensure a sufficient hydrogen mass flowrate to fulfill the hydrogen recombination rate dictated by the correlation.

As already mentioned, stratification phenomena are underlined by the simulation of the recombining located at the elevation of 12m from the bottom. Figure 6 shows the time history of hydrogen distribution inside the volume at different elevations. It can be noted that, during the transient, the hydrogen concentration in the portion of space below the recombining varies much more slowly with respect to the portion above the recombining.

Fig. 7 shows, at the end of one hour of transient, the hydrogen distribution in the space around the recombining, for the three device locations.
Fig. 1 Correlation of reacting hydrogen masses for two recombiners of the Zx series of experiments in the BMC.

Fig. 2 Sensitivity analyses with GOTHIC for a NIS recombiner. Gas inlet flow velocity for device location at 1 m, 3 m, 12 m from the volume bottom.

Fig. 3 Sensitivity analyses with GOTHIC for a NIS recombiner. Gas temperature at recombiner inlet and outlet for a device located at 1 m from the volume bottom.

Fig. 4 Sensitivity analyses with GOTHIC for a NIS Recombiner. Recombiner efficiency for a device located at 1 m - 3 m - 12 m from the volume bottom.
The above results confirm the consistency of the predictions of the GOTHIC code, implemented with the new algorithms simulating a catalytic hydrogen recombiner, and encourage to move to more complex simulations (multicompartment geometries with several recombiners).

4 Hydrogen Mixing and Recombination in a Multicompartment Geometry

4.1 The Battelle Model Containment Simulation
By the Eulerian Code

The Battelle Model Containment (BMC) ([/3]), fig. 8) facility has been used for many years in order to investigate physical phenomena relevant for nuclear power plants containment performance and qualify computer codes.

Experiments have been specifically conducted on catalytic hydrogen recombiners, some of which featured the presence of several devices in different subcompartments.

A 3D BMC - GOTHIC model, developed in accordance with the findings and choices discussed earlier and including the specifically developed recombiner model has been set up.

The BMC is described by means of one single subdivided volume provided with 720 nodes (72 parallel vertical channels with 10 vertical subdivisions). This is considered the minimum number of nodes to correctly capture the fluid motions in the space surrounding the recombiners. Three different recombiners modules (manufactured by NIS and SIEMENS) are simulated: heat sinks are described in detail in order to properly account for their heat removal (and its effect on convective flow patterns). Different rooms have been simulated by the peculiar modelling technique of the GOTHIC code using variation tables in order to model the complex 3D domain, trying to minimize the number of closed cells which would determine a loss of computational efficiency. Gas release (H2 and steam) during the test is simulated by boundary conditions and associated functions.

Figures 9 and 10 illustrate the BMC GOTHIC model, meanwhile figures 11 and 12 show preliminary natural circulation flow patterns predicted with only one (NIS) recombiner model (located in the dome of the BMC) activated.

The work on the detailed simulation of multirecombiner experiments by GOTHIC, as well as the validation of the methodology to tune the GOTHIC models on selected steady state runs of the CFD code CFX 4.1 is currently in progress.
Fig. 5 Sensitivity analyses with GOTHIC for a NIS recombinder. Hydrogen inlet flowrate against recombination rate for device located at 1 m from volume bottom.

Fig. 6 Sensitivity analyses with GOTHIC for a NIS recombinder. Hydrogen concentration at different elevations in the central channel.

Fig. 7 Sensitivity analyses with GOTHIC for a NIS recombinder. Hydrogen distribution at the end of the transient with recombinder device located at 1 m, 3 m and 12 m from the bottom.
Fig. 8 Configuration of the Battelle model containment for the Zs recombiner tests

Fig. 9 BMC GOTHIC Model. XxY Noding

Fig. 10 BMC GOTHIC Model. Elevation Grid
Fig. 11 BMC GOTHIC model. Natural circulation fluid motions sensitivity with only the NIS recombine. Cross section A-A

Fig. 12 BMC GOTHIC model. Natural circulation fluid motions sensitivity with only the NIS recombine. Cross Section B-B

Fig. 13 Vertical (left, R1, R2, R3, R9) and horizontal (right, R3) cuts through the grid of the BMC model
4.2 The Battelle Model Containment CFD Model

In terms of CFD code model the physical and geometrical representation of Zx-experiments should be accurate and detailed to really establish the desired reference flow field. The geometrical model has been created at GRS ([4/]) while the physical description and the analyses will be performed by ANSALDO.

To develop a model of the BMC for CFX 4.1 (see fig. 13) appears to be a rather complex task. In the Zx-configuration length scales to be resolved range from 0.03 m (hydrogen injection pipes) to several meters (in the compartments). These compartments are generally bended and curved. Thirteen flow paths connect the six main compartments. In order to produce a numerically smooth grid and to resolve all gas injection and recombiners locations, a total number of about 97000 cells is currently used. Two cuts are provided in Fig. 13 to give an idea of main parts of the grid in the Battelle Model Containment CFX model. The vertical cut includes the outer compartments and the central room (R1-R2-R3) with all injection locations and one recombiner. The horizontal cut in this figure zooms into this compartment and illustrates how a focusing grid is employed to properly resolve small pipes in order to reduce numerical diffusion.

4.3 The link procedure between GOTHIC and CFX code

The link procedure between the Eulerian computer code GOTHIC and the CFD code CFX4.1 is under development at ANSALDO. The goal of the coupling between the two codes is to extend the applicability of the simulation methodology to a real plant case under severe accident conditions. The coupling is first to be set up and tested for predefined BMC Zx-series experiments. The GOTHIC code is used to simulate the test transient. Then, for selected times into the transient, the GOTHIC solution is extracted and a number of calculated parameters are transferred as boundary conditions to the CFX4.1 for a steady state run. The aim of the CFD steady state run is to validate the GOTHIC global solution (namely the natural convection / hydrogen distribution pattern) for the selected time. The basic assumption for this procedure is that the fluid dynamic transient time scale is very small if compared with the gas mixture composition and wall temperatures transient evolution time scale.

The list of information extracted from the GOTHIC transient solution at a given time and transferred as boundary conditions to the CFX4.1 steady-state calculation is the following:

- wall surface temperature of each wall in each node of the GOTHIC BMC model
- total hydrogen recombination rate (sum of the 3 recombiners)
- total steam condensation rate
- total hydrogen mass in the containment
- total steam mass in the containment
- gas mixture pressure
- gas mixture average temperature in the containment

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The GOTHIC predicted wall surfaces temperatures are assigned to each patch of the CFX model. The GOTHIC gas mixture pressure, volume average temperature and gas composition is assigned as a uniform field in all the CFX domain (initial guess solution).

Since the goal is to calculate a quasi-steady state condition in CFX, a set of mathematical steady-state boundary conditions has to be defined for this calculation. This is achieved by assuming that, at the selected time, the hydrogen injection rate and the steam injection rate are equal to hydrogen recombination and steam condensation rates calculated by GOTHIC, even if there is a mismatch between the quantities in the long term.

The CFX simulation will consider a fully-compressible, multicomponent flow (air, steam, hydrogen). The recombiner model will be included in the CFX model. The recombiner itself is modelled as a solid region and its physical inlet and outlet are the outlet and the inlet of the CFX computational domain respectively. The gas mixture conditions at the recombiners boundaries (temperature, composition and velocity) are calculated via the CFX FORTRAN interface which represents the CFX recombiner model. The activity to tune this linking procedure by benchmarking against Zx- experiments is in progress.

5 Summary and Conclusions

The goal to effectively (i.e. with reasonable computer running time) and accurately (i.e. with a high degree of confidence on the results) predict the natural convection flow patterns and the distribution of steam and gases in the containment of a nuclear power plant following a postulated accident with significant fuel damage in the presence, amongst other heat sinks and sources, of catalytic hydrogen recombiners, has prompted the conception and development of an innovative simulation methodology.

It aims to exploit, to the practicable extent, both the strengths of distributed parameter computer codes, based on the Euler approximation of the momentum equations (reduced computer time) and CFD of codes based on the full Navier-Stokes formulation (accuracy of predictions).

The basic idea is to run the transient simulation with the Eulerian code and to use the CFD code, run at selected times with the steady state option, as a reference code to tune the Eulerian code model.

The methodology development and validation against experiments performed in the Battelle Model Containment is the object of two projects sponsored by the Commission of the European Communities within the IV Framework Program on Reactor Safety. Organizations participating to the project include: Ansaldo and DCMN of Pisa University (Italy), Battelle Institute and GRS (Germany), Politecnical University of Madrid (Spain).
The following activities have been performed or are currently in progress:

(i) experimental tests in the Battelle Model Containment
(ii) addition to the Eulerian computer code GOTHIC of both an empirical recombining performance model and a detailed recombining thermohydraulic model
(iii) simulation of convective motions in a single volume (simple geometry) with the upgraded GOTHIC
(iv) simulation of the BMC experiments with the upgraded GOTHIC
(v) simulation of the BMC facility with the CFD code CFX 4.1
(vi) definition and set up of the link procedure between GOTHIC and CFX 4.1.

Results from activities (ii), (iii) and (iv) are presented in the paper. The upgraded GOTHIC predictions are judged encouraging for the continuation of the projects.

The BMC facility model for the CFX 4.1 code and the link procedure between GOTHIC and CFX 4.1 are currently under validation by using the BMC experimental test results.

Following successful validation the methodology might be adopted for a real containment application since, given the great generality of a CFD code, its accuracy should be practically independent from the problem geometry, provided that the proper noding mesh and the code basic options adequate capture the relevant phenomenologies are employed.

References


ANTHEM2000™: Integration of the ANTHEM Thermal-Hydraulic Model in the ROSE™ Environment

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Abstract

ROSE™ is an object oriented, visual programming environment used for many applications, including the development of power plant simulators. ROSE provides an integrated suite of tools for the creation, calibration, test, integration, configuration management and documentation of process, electrical and I&C models.

CAE recently undertook an ambitious project to integrate its two phase thermal hydraulic model ANTHEM™ into the ROSE environment. ANTHEM is a non-equilibrium, non-homogenous model based on the drift flux formalism. CAE has used the model in numerous two phase applications for nuclear and fossil power plant simulators.

The integration of ANTHEM into ROSE brings the full power of visual based programming to two phase modeling applications. Features include graphical model building, calibration tools, a superior test environment and process visualization. In addition the integration of ANTHEM into ROSE makes it possible to easily apply the fidelity of ANTHEM to BOP applications.

This paper describes the implementation of the ANTHEM model within the ROSE environment and gives examples of its use.

1. Introduction

ROSE [1], an acronym for Real-time Object-oriented Simulation Environment, is an component based, visual programming environment for the development of simulation models. ROSE combines powerful development, test, documentation and system integration tools and a user-friendly Windows/Motif compliant GUI into an integrated model development environment. ROSE is applicable to the entire simulator life cycle, from model design through to testing, documentation and long term maintenance.

ROSE has been used in many simulation or design applications and is currently in use in several power plant simulators. The scope of ROSE modeling typically includes all electrics, I&C and homogeneous flow modeling.

ANTHEM is a non-equilibrium, non-homogenous model based on the drift flux formalism. The addition of ANTHEM to the ROSE modeling suite, coined ANTHEM2000™ is possible because of the generalized nature of the model, its overall maturity and the architecture of ROSE itself. The project has
been motivated by the following considerations:

- providing CAE customers the power of a ROSE-based solution for NSSS applications.
- reducing development costs for NSSS simulation applications by providing CAE developers a graphical model building capability and a graphical testing environment.
- facilitating visualization of the complex phenomena of two phase flow.
- improving data traceability for NSSS models.
- providing improved fidelity for BOP applications.

The latter is particularly important. ANTHEM2000 will make it possible for CAE to meet the increasing industry demand for two phase modeling capability for BOP systems within the overall ROSE environment. Simulation of such phenomena as feedwater line boiling, steam line filling, boiling heat transfer in heat exchangers and system drain down for BOP systems is possible. In fact ANTHEM has already been used to simulate feedwater and RHR systems.

This paper describes the key features of ANTHEM2000. The following section provides an overview of the ANTHEM. Section 3 provides an overview of ROSE. Section 4 discuss issues specific to the integration of ANTHEM and ROSE. Section 5 provides a summary and conclusions.

### 2. ANTHEM Overview

ANTHEM was originally developed by CAE to provide a high fidelity two phase thermal-hydraulic model of the NSSS and steam generators for power plant training simulators. ANTHEM has been used for the full range of simulator NSSS applications from mid-loop operations to severe transients leading to fuel failure and hydrogen generation. ANTHEM has been or is being installed in 15 full scope simulators or simulator upgrades supplied by CAE.

ANTHEM is a generalized non-equilibrium, nonhomogeneous drift flux model of two phase flow. The basic thermal-hydraulic model uses six conservation equations: conservation of liquid mass, gas mass and non-condensible mass, conservation of liquid energy and gas energy and conservation of mixture momentum. The thermal-hydraulic model uses a staggered grid approach in which the mass and energy equations are solved for each control volume and the momentum equation is solved for each flow path. Phasic flow rates are determined from the drift flux correlation and the mixture momentum equation. Convected quantities required for the resolution of the phasic flow rates are determined using an upwind donorig scheme. The model solves for pressures based on a local momentum equation. Neither the loop momentum nor the local pressure approximations are used.

ANTHEM uses a one step, non-iterative, semi-implicit, Courant violating numerical algorithm. This algorithm provides great flexibility with respect to the density of nodalization. Numerical drift is controlled by applying an error term to the mass conservation equation that is calculated as the difference between an integrated density and the density calculated from the equation of state. The error term is applied locally and for each phase.

ANTHEM has been used on PWR, BWR and fossil simulators. It is typically coupled to CAE’s neutronic
model COMET. COMET provides local thermal power to ANTHEM, while ANTHEM calculates local fuel temperature and moderator properties. Validation of ANTHEM [2,3,4] and COMET [5,6] has been previously reported.

3. ROSE Overview

The basic components of the ROSE toolset are:

- Graphics Editor. The graphics editor is used to create object icons and specify associated dynamics.

- Object Editor. The object editor is used to create the object code and variables associated with the icon and to specify data flow between connect points.

- Model Editor (me). The model editor is used to design the simulation model. A ROSE model is easily built by dragging objects representing physical devices from the object libraries onto a schematic, entering data and making the appropriate connections. Both Model Editor and Object Editor are supported by extensive on line library documentation. The documentation is in HTML format and can be viewed by standard commercial browsers.

- The builder allows the user to define, compile and link test configurations by graphically extracting schematics from the database repository and combining them into entry points, specify calling order and iteration frequency. The builder invokes the code generators appropriate to the physical system being modeled. The code generators include hydraulic, electrical, discrete and analog logic code generators. The builder can also be used to link manually coded source files to the configuration.

- Runtime. Runtime allows the user to initialize and control simulation, manipulate the simulated system through mimic panels or object touch areas, visualize the results, examine and plot individual variables. Interfaces to 3D visualization and multi-media effects are also supported.

- Document Generator. Information about objects may be extracted from the ROSE database using and Object Query Language (OQL). Users can use templates or develop special purpose tools to access the database and generate reports.

- Site integration tools. Site integration tools provide configuration control required for large complex, multi user projects.

Configuration control, one of the most important aspects of ROSE, is also among its least visible. ROSE has it origins in internal software productivity tools intended to standardize model design and optimize simulator development. Thus, ROSE is designed to support a single user on a stand alone workstation or a multi-user, multi-project environment including shared development between remote sites [7]. A data repository is used to store all object data and make this data available to all tools and applications. Schematics are checked out to local clients for development. Checked out schematics are write locked to prevent multiple write access. In addition the entire database can be copied to a standalone client
database. Multiple client databases can be created to support individual testing or multiple simulator configuration loads.

4. ANTHEM-ROSE Integration

4.1 ANTHEM Two Phase Library and Code Generator

The integration of ANTHEM into ROSE required the creation of a new code generator and the two phase object library. One of the original fundamental design criteria for ROSE was the requirement that the ROSE architecture easily support the incorporation of new model classes. The ROSE API effectively hides the database implementation and HMI from the code generator developer. In order to understand how this is achieved it is important to understand the interrelated role played by code generators and objects.

A ROSE object is defined through the Object Editor. The object definition consists of the graphical icon and its associated dynamics, model variables and associated units, description and dataflow, connect point definition and object code. Connect points specify what variables are passed between objects when they are connected. Multiple connect points with different variable lists are supported. The model code can be implemented as a language independent pseudo code and/or as calls to external subroutines. The latter facilitates encapsulation of existing, manually coded models into ROSE objects. For example the objects in the two phase library for four quadrant recirculation pumps, jet pumps and relief valves all use subroutines previously validated with the conventionally coded two phase model.

ROSE has two classes of code generators: the sequential code generator and network code generators. In many practical applications both types of code generators are invoked by the builder. The sequential code generator translates the object pseudo code for the object instances on the schematic to the target simulation language. Shared memory variable names are created according to the dataflow specification. Thus, the sequential code generator is used for discrete, stand alone components such as pumps, valves and controllers where an explicit approach is appropriate. The sequential code generator calls objects in the order that they are instantiated. However, the user can reorder the call order without physically changing the schematic.

An object definition does not require code. This is the case for objects that are typically associated with networks, such as nodes and links - the actual model is implemented by the network code generator. Network code generators are purpose built code generators that are invoked when an implicit numerical solution for the particular physical process is required. The typical applications are hydraulic networks, electrical networks and relay networks. In practice, a network related object can also include pseudo code for those portions of the model that can be treated explicitly. The choice is largely a matter of convenience. For example, the node and link objects in the ANTHEM library include auxiliary scaling calculations to drive the object dynamics that are implemented as pseudo code.

The ANTHEM network code generator consists of two parts: “netlist” generation and code generation. The netlist provides the code generator basic nodal connectivity information and identifies objects of
concern to the code generator. The required information is extracted from the schematic by querying the database. The code generator itself creates a specific instance of the model code. As such, it performs three functions: it creates data tables that contain the network mapping information, it creates the mapping information between external boundary conditions and the model and it invokes the various generalized subroutines that constitute the model itself.

### 4.2 Object Calibrators and Data Referencing

ANTHEM is a complex model. However since the model itself is mature and has been generalized for arbitrary configurations, any specific application is data driven rather than code driven. Data identification, entry and tracking are thus issues of prime importance. ROSE facilitates this task through the use of Object Calibrators[8] and Calibration Notes.

In the model editor the "informat" provides information about all variables associated with a variable. The Object Calibrator is invoked from the informat. The Object Calibrator is a user-friendly graphical, interactive tool which provides a standardized approach to calibrating objects. The Object Calibrator performs two basic functions: it identifies the required input data, with their units, necessary to calibrate a particular object; it converts the user-supplied input data into units that are compatible with the model code and then calculates all the parameters/ constants required for the simulation.

![Figure 1: Node object calibrator](image)

Calibration Notes provides a mechanism of tracking reference information associated with the data. Calibration Notes allow the user to provide data references and other documentation information for the calibrator input - such as offline calculations for the estimated volume of a particular control volume or the correlation used for a particular form factor. All information entered through the Calibration Notes
window is stored in the database and tagged to the object instance. Thus all documentation information relevant to a particular model can be stored on line and retrieved by ROSE’s object query tools. Figures 1 and 2 show the Object Calibrator for node objects and a corresponding Calibration Note. Used together the Object Calibrator and Calibration Notes facilitate model initialization and data traceability.

\[ \text{Core Diameter} \ D = 11.06 \ \text{ft} \]
\[ \text{Fuel rod diameter} \ d = 0.374 \ \text{in} \]
\[ \text{Number of fuel rods} \ N = 58852 \]
\[ \text{Core cross-sectional area} = \pi(D/2)^2 = 96.81 \ \text{ft}^2 \]
\[ \text{Flow area} = \pi - (D/2 - (N/4)^2) = 57.2 \ \text{ft}^2 \]

Figure 2: Calibration Notes

4.3 ROSE Runtime and Visualization

As stated earlier ROSE run time provides the means to test and visualize the simulation. The standard run time features include the following:

- the infomatrix, which provides a continuously updated readout of all object variables in tabular form and a means to change the value of any simulation variable.
- the inspector, which allows the user to monitor in tabular form a personal selection of variables from a variety of objects.
- the network object, which allows the user to monitor all variables of a particular type (all pressures for example) from all schematics in the group.
- Autograph™, which allows the user to plot any variable against time. Plots can be saved to a file for later analysis or overlaid against previously recorded data. Out of tolerance data is automatically indicated.
- readouts, which provide an 'at a glance' display of important parameters.
- touch areas to control device state (pumps, motors, valves, relay etc)
- object dynamics, used to visualize system behavior

All ROSE schematics can be much larger than the physical screen. Thus, pan and zoom are incorporated to allow the user to focus on a particular area of interest.

An example of a schematic used to model a Westinghouse four loop RCS is shown in figure 3. Object dynamics include:

- readouts of state variables on nodes.
- dynamics to indicate flow direction.
- variable color background to indicate level.
- color dynamics to indicate the logical state of devices such as pumps, motors, valves and switch positions.
- color dynamics on nodes to indicate spatial variation of temperature and void fraction across the network.
- color dynamics on pipes to indicate flow rate.

The upper and lower limits represented by a set of colors can be reset to allow the user to choose the resolution appropriate to the problem.

Object dynamics are created through the graphic editor. The choices for visualization are numerous and are only limited by the imagination. Other possibilities include: different shaped node objects to better reflect the actual shape of control volumes, color dynamics to indicate the degree of thermal nonequilibrium, phase flow, phase mass, fuel radial temperature, bar charts, embedded plots, meters and the dynamic application of texture.

The design of a particular visual representation is largely subjective and is dependant to a large extent on the goals of the user. Thus, the most important aspect of a visualization tool is not the value of any particular display but the flexibility of the underlying graphic editor to support a large number of different possibilities. The underlying ROSE graphic editor was originally designed to support mimic panels and hence supports a large number of dynamics.
5. Validation

Validation and testing of ANTHEM2000 is currently underway. The purpose of the validation is to ensure that the ROSE generated models behave as expected compared to the manually generated versions of ANTHEM. Thus, validation consists primarily of comparing ANTHEM2000 test results against previously delivered versions of ANTHEM currently being used for training. The current effort is concentrated on a comparison of an ANTHEM-2000 simulation of the South Texas Project (STP) four loop Westinghouse NSSS. The STP simulator was delivered by CAE in 1995. Validation includes repeating all ANSI 3.5 (1993) Appendix B transients and normal operations. In addition experimentation with different visual representations is underway.
6. Conclusion

The integration of ANTHEM into the ROSE environment provides a powerful tool for the simulation and visualization of two phase flow phenomena for power plant training simulators. ANTHEM provides a validated model while ROSE provides a graphical model building tool that can be used to design, calibrate, test, document and maintain the model during its entire life cycle.

References


ROSE, ANTHEM, ANTHEM2000 and AUTOGRAPH are trademarks of CAE Electronics Ltd.
The Implementation of a Mid-loop Model for 
Doel ½ Training Simulator

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mid-loop, model, simulator

ABSTRACT

To cope with upgrade requirements of the Full Scope training simulator of Doel 1/2
(Belgium), a 5-equation model has been implemented for mid-loop operation training.
This model will permit to simulate the following conditions:
• Normal operating conditions
  Draining of the primary circuit at vacuum conditions
  Venting of the primary loop with the help of a vacuum pump
  Filling-up of the primary circuit
• Incident and Accident conditions
  Loss of RHR (Cavitation of RHR pumps)
  Reactor heat-up and boiling

In order to simulate the pressurizer water hold-up and loss of steam generator reflux
cooling, flooding correlations are used predicting steam generator U-tube and
pressurizer surge line flooding. Loss of horizontal stratification in the hot leg has been
taken into account. A steam generator piston model for heat transfer has been
implemented.

This paper describes the mid-loop model specifications, its implementation and testing
in the simulator environment. Special attention is given on how the model has been
integrated within the existing simulator.
1. Introduction

The major concerns following the loss of residual heat removal (RHR) system events during midloop operation are well known. Due to the reduced water inventory, the core can heatup quickly and start to boil. If no action is taken, boiling can lead to core uncovering in a relatively short time [1].

Another important characteristic is the low primary pressure (atmospheric and even vacuum for the Doel 1/2 power plant). Even at greatly reduced nuclear power and consequently low flowrates, the very low steam density leads to high steam velocities. At these high velocities, flooding phenomena can occur, resulting in [1] [3]:

- loss of reflux cooling in the steam generators;
- water hold-up in the SG inlet plena;
- water hold-up in the pressurizer.

All those effects can result in a higher primary pressure, which in turn leads to:

- faster core depressurization and uncoverage;
- damage to seals such as the SG nozzle dams;
- loss of gravity drain from the Reactor Water Storage Tanks.

The final aspect is the specific plant configuration. During midloop operation the pressurizer vent valves or PORVS can be opened, the manway on the pressurizer may be open, the SG nozzle dams can be installed, and, last but not least, workers can be present in the primary containment.

2. Mid-loop model specification

At the delivery of the Doel 4 full scope simulator, a simplified non-condensible model (based on the hydrostatic equilibrium in a coarse nodalization) was integrated in the reactor coolant system (RCS) model. Although this simplified model satisfies the training needs for normal operation conditions, it cannot treat reactor heat up and boiling with a reduced water inventory. Due to the specific draindown and venting procedures of the Doel 1/2 power plant, neither the original RCS model, nor the Doel 4 simplified non-condensible model is adequate for the training needs, even at the normal operating conditions during refueling.
In 1995, it was decided to extend the existing RCS model with a new model for mid-loop operation training. The specifications for this new model are:

- A correct simulation of the normal operating condition including:
  - the draindown from solid state to the mid-loop state at vacuum conditions (by draining from the RHR with all vents closed);
  - the aeration of the primary circuit;
  - the venting of the circuit by means of a vacuum pump;
  - the filling up of the circuit.

- A correct simulation of accident conditions, including:
  - the cavitation of RHR pumps by vortexing at the suction line;
  - core heat up and boiling, up to core uncovercy;
  - pressurizer water hold-up and flooding phenomena in the pressurizer surge line;
  - loss of reflux condensation and flooding phenomena in the SG U tubes;
  - safety injection and return to normal operating conditions.

- The easy and fast implementation of the model in an existing code:
  - easy switch over from the RCS model to the extended model;
  - no interference of the 2 models.

Plant data and instructors experience should guarantee a good response of the model in normal operating conditions. The model response in accident conditions would be tested by engineering calculations and literature results.
3. Model Description

3.1. Basic equations

The model uses the node-flowpath network represented in figure 1.

Each node encloses a control volume representing the fluid mass and energy. The flowpaths connecting the nodes represent the fluid momentum.

The major control volumes are:

1. the inner vessel, upper head and hot legs;
2. the pressurizer and pressurizer surge line;
3. the steam generators;
4. the cold legs, reactor down-comer and lower plenum.

All nodes are treated in thermal nonequilibrium between a mixture phase (water and steam bubbles) and a gaseous phase (steam and non-condensible gas each evaluated at their partial pressure). The conservation equations for each phase are used on every node, with a simplified conservation equation of non-condensibles energy.

On every flow path a single conservation equation of fluid momentum is used except for the loop seal, where an orifice type equation is used (the reactor coolant pumps cannot be started at mid-loop operation).

3.2. Constitutive equations

The homogenous flowrates, resulting from the momentum equation, are separated into liquid and gas flowrates by means of drift velocity correlations and the local fluid conditions at each path end. In order to simulate the water hold-up in the pressurizer, a Kutateladze-type flooding correlation is used [1]. The loss of reflux condensation is represented by a Kutateladze-type flooding correlation in the SG U tubes, the Taitel-Dukler correlation is used to predict the loss of horizontal stratification in the hot legs [1].

Heat transfer in the core is simulated by the existing RCS model using calculation results from the mid-loop model. A piston model [1] for the steam generator heat transfer results in a correct calculation of the active condensation surface of the fresh
steam in the SG U tubes. This model is based on the compression of the non-condensible gas in the lower part of the steam generator and the outlet plenum (at secondary side temperature) by steam produced in the core.

The steam saturation temperature for the wall and mixture interface heat transfer calculations is based on the piston model. Each node is divided into an active (steam at total pressure and corresponding temperature) and an inactive (steam at partial pressure and corresponding temperature) volume, heat rates are calculated with both volumes. Heat transfer to the non-condensible gases is simulated by calculation of the equilibrium temperature (water, wall and steam) and the use of a time constant.

The simulated interface exchanges are:

- condensation at mixture interface (piston model);
- condensation on the walls (piston model);
- evaporation at mixture interface;
- bubble rise;
- droplet fall.

All the existing malfunctions (breaks in the primary loops, steam generator, pressurizer) of the main RCS model, can be activated in the extended model. Break flowrates are calculated using orifice type equations.

3.3. Numerical methods

The linearized conservation equations are integrated by a Runge-Kutta integration method, with a reduced time step (4 iterations per calculation cycle of 125 ms), resulting in a high numerical stability.

4. Model Implementation

One of the major concerns for extending the Doel 1/2 RCS model, was the limitation at a minimum of the existing subroutines to be modified. Figure 2 gives a good representation of the actual RCS model and its communication with interfacing systems. The modifications in the simulator RCS model are:

- manual switch-over between the main RCS model and the model for mid-loop operation training
• setting a flag in all the RCS subroutines (with exception of the core heat transfer subroutine) to freeze them when the mid-loop model is activated;

• modification of the 3 interface subroutines to transmit the appropriate parameters (calculated by the mid-loop or main RCS model) to the interfacing system and the core heat transfer subroutine.

The manual switch-over was chosen to avoid any interference or regression of the main RCS model. On the other hand the mid-loop model should respond correctly to wrong operators actions leading to pressurization of the primary circuit.

5. Model Testing and Results

5.1. Normal Operating Conditions

Normal operating conditions, such as draindown from solid state, aeration, venting by means of a vacuum pump and fill up of the primary circuit to cold shut down, were compared with plant data and instructors experience. Pressures, level and temperature evolutions correspond very well, both on a qualitative as on a quantitative base.

As an illustration, figures 3-5 give the result of a controlled draindown (20 kg/s and all vents closed) followed by the ventilation of the primary circuit.

The initial conditions for this event are:

• pressurizer level at 50 %, mixture temperature of 90° C, giving a gas space with 25 % non-condensible;

• primary loops solid : hot leg temperature of 37° C, cold legs at 26° C.

The mixture levels in figure 3-4 illustrate the voiding at different RCS locations. Voiding of the steam generators is initiated at 750 s, when the local pressure reaches the corresponding saturation pressure. Once the pressurizer surge line level reaches the hot leg connection, non condensible gas can enter the steam generator upper parts. The cold part of the primary circuit (downcomer and cold legs) starts voiding when the gas bubble in the core upper head reaches the downcomer bypass. After the mid-loop situation was reached (at 4800 s), the draindown was stopped and the vents to the loop seal were opened, resulting in the pressure rise of the primary circuit shown in figure 5.
5.2. Accident Conditions

The following 5 events were used to validate the model at accident conditions:

- Loss of RHR-cooling by a break at the pumps discharge;

- LOCA in a hot leg, stabilisation at a feed (safety injection) and bleed (hot leg break) cooling mode;

- Complete loss of RHR-cooling, with a large vent path on the pressurizer, steam generators nozzle dams installed -> pressurizer flooding;

- Complete loss of RHR-cooling in a closed primary circuit with both steam generators available -> steam generator reflux cooling mode with a maximum primary pressure of 6 bars.

- LOCA in a cold leg, reactor boiling up to core uncovery.

At the absence of RELAP test results, these events were evaluated on a qualitative base only, using engineering calculations, and literature results [1], [3]. As an illustration, the events 1 and 3 will be discussed here.

5.2.1. Break at RHR-pump discharge event

The loss of primary coolant (10 kg/s) and the resulting hot leg level decrease (fig. 6), leads to vortexting at the RHR-pump inlet, after 2 min., illustrated by the RHR-flow oscillations shown in figure 7. Once the hot leg level reaches the pumps suction line (T₀ + 6 min), all reactor coolant is lost and the primary circuit heats up (± 4°C/min) (figure 8). At time T₀ + 10.20 min, the fuel rods and cladding reach 100°C, a pool boiling regime is installed, and the heat up gradient increases to ± 5°C/min, conform the remaining primary inventory and the residual heat of 8 MW. During reactor heat-up, temperature and resulting density differences between the reactor and the cold downcomer, result in the increasing level difference between the cold and hot legs (fig. 6).

At T₀ + 14.30 min, a safety injection is started to restore hot leg level to the minimum value for RHR-cooling. As this level is reached at T₀ + 18.20 min., the non-affected RHR-train is started, the reactor is cooled down to its initial temperature level, and safety injection is stopped.
This event gives a good insight in the vortex phenomenon at the RHR-pump inlet and the erroneous hot leg level indications during reactor heat-up.

Reactor heat-up gradients are fully consistent with the residual heat and remaining primary inventory.

### 5.2.2. Loss of RHR with a large vent path on the pressurizer

In order to evaluate the model on a more quantitative base, the response to this event was compared with a Westinghouse study using the WGOTHIC code for the 2 loop Mid-loop Case, [3].

The initial conditions for this event being:

- hot leg level at 50% of surge-line section;
- residual heat of 7.3 MW;
- pressurizer relief path of 0.0447 m² [open PORV’s 1];
- hot and cold leg temperature at 37.8°C;

the core heats up quickly and starts to boil ± 650 s [1080 s in the reference study] after shut-down cooling is lost. On the other hand the primary pressure increases [fig 9], leading to a gas release through the open PORV’s. At the beginning, the vapour flow rate to the pressurizer is low enough that the entrained water can fall back through the surge line in countercurrent flow, resulting in the surge line liquid flow oscillations. After ± 800 s [1200 s] however, the vapour flow rate to the pressurizer reaches its flooding limits and saturated water is forced into the pressurizer surge line [fig. 10].

This additional head (water level in the pressurizer, pressure losses in the surge line) results in a peak pressure on the steam generator nozzle dams of 1.9 bar versus 1.55 bar in the reference study. This difference is related to the coarse nodalization of the mid-loop model: the surge line being part of the pressurizer node, it will be filled with saturated water giving an additional water column of ± 2 m [fig. 11]. During the rest of the event, steam is constantly relieved through the opened PORV’s, resulting in core uncovering 2 hr 25 min after the event was initiated (versus 1hr45 min in the Westinghouse study). This difference can be explained by the core configuration of the DOEL 1 simulator: the top of the fuel rods are situated ± 2 m lower than in the reference configuration.
Another major difference is the faster down comer heat-up in the mid-loop model, due to additional heat transferred through the reactor structures, which is not implemented in the Westinghouse configuration, and a higher vapour flow through the reactor by-pass.

Despite the differences of maximum nozzle dam pressure and time to core uncoverey, it can be concluded that the model gives a good qualitative insight in phenomena like pressurizer surgeline flooding.

However, to evaluate the model response on a quantitative base, comparisons with more detailed and personalized studies should be made.

6. Conclusions

The RCS model of the Doel 1/2 simulator was extended with a model for mid-loop operation training purpose. The model response at normal operating conditions is fully consistent with instructors experience and plant data. The model gives a qualitatively good insight for phenomena like surgeline flooding and pressurizer water hold up during accidents where boiling of the core cannot be avoided. Further testing should be done to evaluate quantitatively the model response. The model can easily be implemented in the existing RCS model, with a limited subroutines to be modified.

References


Figure 1
Node - flowpath of the Doel 1 power plant

Figure 2
DATA FLOW representation
Figure 3
Draindown: pressurizer level

Figure 4
Draindown: mixture levels

Figure 5
Draindown: core and pressurizer pressure
Figure 6
RHR- Discharge break: hot and cold leg level

Figure 7
RHR- Discharge break: RHR-flowrates

Figure 8
RHR- Discharge break: hot and cold temperature
Figure 9
Loss of RHR with large vent path: pressures

Figure 10
Loss of RHR with large vent path: surge line flow

Figure 11
Loss of RHR with large vent path: liquid levels
The "IMPACT Super-Simulation" Project for Exploring NPP Fundamental Phenomena

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ABSTRACT

To ensure the safe operation of a nuclear power plant under any normal operating or hypothesized accident conditions, understanding of the physical phenomena involved and tools to calculate such events must exist. The complexity of the fundamental phenomena and their interrelated effects require considerable research and development of software and hardware to produce a credible power plant simulation system. A large-scale project is underway at the Nuclear Power Engineering Corporation to develop the software for a "Super Simulation" system, capable of analyzing scenarios ranging from normal operation to hypothesized accident conditions. Designed as a large-scale system of interconnected, hierarchical modules, IMPACT's distinguishing features include mechanistic models and high speed simulation on parallel processing computers. The present plan is a ten-year program starting from 1993. The IMPACT software will be completed within the year 2000 followed by two years for refinement through extensive verification and validation against test results and available real plant data. Parallelization was completed for the incompressible single phase flow module and on increase in calculation speed of at least 30 times by parallel processing on 64 processors has been attained depending on a number of mesh. The pre-mixing sub-module for analysis of steam explosion phenomena under severe accident conditions has been completed, and was shown to simulate the MSHA tests well. Calculation results of debris spreading model in debris cooling process are compared and showed good agreement with the UC Santa Barbara test results. The physical models in the Boiling Transition Code has been completed and partially verified for both BWR fuels and PWR fuels by the comparison with basic experimental results.

I. INTRODUCTION

Since the accidents at the Three Mile Island-2 and Chernobyl-4 nuclear power plants, the international demand for verification of the safety of power plants has increased. The range of conditions which must be considered now extends beyond the traditional scope of design basis accident conditions to include hypothesized accident conditions. In addition to fluid flow and heat transfer events occurring under design basis accident conditions, phenomena to be evaluated include fuel melt and relocation, emission and transport of radioactive fission products, response to loads on the containment vessel, numerous other events, and associated coupling effects. Because these phenomena are difficult to investigate experimentally, analytical evaluations of the safety margin must be relied upon. Computer simulations based upon fundamental physics principles and sophisticated modeling technologies are required, to satisfy these analytical needs. In addition, visualization techniques for displaying the results must be developed, to assist in comprehension of such complicated phenomena.

IMPACT is the name of a program and of specific "Super Simulation" software, which will perform full-scale, detailed calculations of physical and chemical phenomena in a nuclear power plant for a wide range of scenarios [1-2]. The Nuclear Power Engineering Corporation undertook management of the program and detailed the conceptual plan during fiscal 1993, with financial sponsorship from the Japanese government's Ministry of International Trade and Industry. IMPACT was formulated into a ten-year program, divided into an initial one-year planning phase followed by three technical development phases. An acronym for "Integrated Modular Plant Analysis and Computing Technology", IMPACT's success is anticipated to have a large impact on nuclear power plant safety research and simulation technologies.

II. OVER ALL DESIGN

A. Unique Features

The basic policy was first established to develop the
IMPACT software system for analysis of the available safety margin under a wide spectrum of scenarios ranging from normal operation to hypothetical, beyond-design-basis-accident events. The key features are as follows:

1. Minimization of use of empirical correlations and constants,
2. Maximization of use of mechanistic models and theoretically based equations,
3. Utilization of a parallel processing computer, (IBM Power SP-2 with 72 processors and maximum total performance of 18.7 GFLOPS),
4. User assistance from input generation to comprehension and retention of results,
5. Modular structure for easy development and maintenance, and
6. Fast running analysis modules for parametric surveys and detailed analysis modules with high-accuracy for verification of safety margin.

B. Development Plan

IMPACT was formulated into a ten-year program. To justify continuation of the long term project, it is important to provide check points for verification of the results obtained. For this purpose, the program was divided into an initial one-year planning phase followed by three technical development phases of four-, three-, and two-years in length, as shown in Table 1.

Table 1. Development Schedule (Fiscal Year)

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Phase-1 is a period of technology acquisition for developing parallel computing software and understanding detailed phenomena to plan for incorporation of mechanistic modeling. It will be demonstrated at the end of Phase-1 that high-speed and large-scale calculations with existing mechanistic models can be executed on the parallel processing computer. For this purpose, the Boiling Transition Code and the Flow Induced Vibration Code are now under development within the IMPACT program. Since many of the mechanistic models for these phenomena exist, efforts can be put into effective parallel processing. These codes will be completed within Phase-1. The third major code in the IMPACT system, a prototype version of Severe Accident
Code, will be completed during Phase-1. The prototype version will consist of fast-running analysis modules with models which are mainly experimentally based. As for the Control System, Simulation Supervisory System to integrate fast-running analysis modules will be developed.

Phase-2 is a period of software enhancement. The detailed analysis modules for the Severe Accident Code will be completed within Phase-2. The experimentally based models in the fast-running modules will be supplemented by mechanistic models to enable safety margin calculation. For phenomena which are not yet fully understood, structural and physics models will be selected so as to rely as little as possible on experimental data. Increase in analysis accuracy will be attempted for better quantification of safety margin and for optimization of accident management procedures. In addition, extensive parametric surveys will be performed using these fast-running modules, for various event sequences to better define the severe accident scenarios. The Boiling Transition Code and the Flow Induced Vibration Code will be transferred to users for application. The Control System, which consists of the Simulation Supervisory System, Information Management System, Data-base Management System, Human Interface System, etc. will be completed.

Phase-3 is a period of user environment enhancement. This includes improved visualization. The Severe Accident Code will be put to practical use and be brought to maturity through various applications.

III. IMPACT SYSTEM STRUCTURE

The groups of modules shown in Fig. 1 will make up the IMPACT simulation system. Three of the main module systems are described as follows.

![Diagram of IMPACT System Structure]

Fig. 1 Ultimate Software Structure of IMPACT
A. Control System

The various functions in the IMPACT simulation will be performed by independent groups of modules, or "systems". These systems will be managed by independent control systems which also consist of several modules each. The IMPACT Control System will integrate all of these.

The major portion of the Control System, the Simulation Supervisory System will manage the Analysis System. It will call and terminate analyses and input/output modules as appropriate, with respect to time in the event and to physical location, as shown in Fig. 2. The system will enable modules to run in parallel or time-sequentially. With these capabilities, it will make possible integration of multiple modules for unified, efficient calculations. The system will coordinate time stepping in the various modules, which have their own time step sizes, and have capability of parallelization control, that is dynamic allocation of processor elements to optimize simulation elapsed time.

The Control System will also be able to retrieve and distribute information necessary for simulation in full scope. It will obtain information from the input data, data bases, and knowledge bases including simulation conditions, accident scenarios, and various criteria such as conditions for initiating safety devices. Based on these data, the system will call analysis modules and input/output modules, etc. Scenarios and various criteria will be recorded without being programmed into the code and will be able to be changed freely. Such control and management will be possible even when the various modules are executed in parallel. Consequently, the control system manages multiple modules while appearing to the users to be a single code.

B. Human Interface System

The Human Interface System supports the Analysis System and the Control System. It will assist with automatic mesh generation and include expert systems which can integrate data from several information management systems and from previous numerical simulations. As examples, boundary and initial condition data may be derived from results of previous simulations and material properties may be obtained from information banks, all of which have been stored in the Data Base. To support the Control System, the Human Interface System will assist in creating scenarios by scheduling operating directives, operator actions, malfunctions, and other events.

C. Analysis System

The IMPACT system in Phase 1 consists of the 3 major codes. In Phase 2, the Analysis System will be a highly flexible system, capable of simulating any plant component or system at any time during an event controlled by the Control System, due to the enhancement of modular structure
for each function or model. This flexible structure will also readily accommodate changes in the calculational degree of detail, function and range of scale.

The modules have been divided into two broad groups, namely, "basic physics modules" and "phenomenon-specific modules". The former modules will analyze basic physics phenomena, such as three-dimensional single- and multiphase flow, which will appear not only in a specific plant component and specific event but also in various components, accident scenarios, and events. The latter modules will treat phenomena specific to accident scenarios such as debris relocation, fission products release, steam explosion, etc. These modules are described in Fig. 2. To reconcile the calculation speed and the degree of detail in the model, detailed analysis modules and fast-running analysis modules, in both module groups will be developed.

(a) Fast-running Analysis Modules
- Philosophy of "The faster-running the better"
- Extensive parameter survey for various event sequences to estimate phenomenological uncertainty bands
- Two classifications: "basic physics modules" and nuclear power plant "phenomenon-specific modules"
- New methods and/or models development or modification
- Major effort to furnish all equipment models required for complete analysis of scenarios
- Parallelization attempts
- "Detailed analysis modules" developed on this modular structure

(b) Detailed Analysis Modules
- Philosophy of "Reality is the Power"
- Increase in module analysis reliability
  -- safety margin quantification
  -- accident management procedures optimization
- Two classifications: "basic physics modules" and nuclear power plant "phenomenon-specific modules"
- Minimal reliance on experimental correlations or constants (mechanistic modeling)
- Three-dimensionalization as much as possible
- Calculation of inter-related phenomena via linkage of separate modules, e.g. fluid, thermal, structural

IV. CURRENT STATUS AND RESULTS

In April, 1994, the IMPACT program began the initial four-year technical development phase. NUPEC has initiated the technical development and is contracting work out to IMPACT team members. Software is being developed on an IBM POWER SP-2 parallel computing system having 72 processor elements with total performance of 18.7 GFLOPS, which was installed in October, 1995.

A. Basic Physics Modules

Current emphasis is put on development of three dimensional single-phase and two-phase flow analysis modules as the basic physics modules. These are being incorporated into the Boiling Transition and Flow Induced Vibration Codes as sub-modules. These flow analysis modules are based on the a-FLOW code [3] for single processor, vector computers.

Parallelization was completed for the incompressible single phase flow module, for analysis in cartesian, cylindrical and boundary-fitted coordinates, and for two-phase and mixed-phase flow modules. In parallelization, a domain decomposition method was used with parallel block ordering in the preconditioning. The benchmark problem was solution of discrete Poisson equations over a large number of mesh, because it requires considerable execution time.

The parallelization effect was evaluated by calculation of three-dimensional flow through an elbow pipe using boundary-fitted coordinates. Calculation speed up is attained, as shown in Fig. 3. The speed up ratio is the ratio of the calculation time on the base number of processors to that on n processors. For a large number of meshes, calculation could not be performed on a single processor due to memory limitation, so the base number of processor is greater than one. Considering this adjustment, speed up ratios of greater than 30 may be claimed for the large mesh, large processor calculations.

The microscopic models such as Molecular Dynamics and Cellular Automata are currently not included in the IMPACT software, since their application to the large scale system of nuclear power plants and reliable physical interpretation of their calculation results has not yet been demonstrated. The applicability of Lattice Gas Automata, Lattice Boltzmann-BGK method, and other mesoscopic-level methods to the IMPACT simulation software is currently under discussion. In the near future, they will be used as standards for verification of empirical models.
B. Phenomenon-Specific Modules

The fast running analysis modules are now in the stage of coding. They will be combined into a single prototype version of the Severe Accident Analysis Code, which can analyze overall behavior of nuclear power plants under severe accident conditions in a once-through manner.

As one of detailed analysis modules, the steam explosion analysis module VESUVIUS is being developed. The two-phase flow analysis module, one of the basic physics module, was extended to handle four-phases; water, steam, particle molten corium, and continuous phase molten corium.

The phenomena of steam explosion can be divided into four stages; Pre-mixing, Triggering, Propagation, and Expansion [4]. The Pre-mixing sub-module with a particle breakup model and a jet breakup model [5] has been completed. The calculation results for particle breakup and mixing are compared in Fig. 4 with the MIXA test results [6]. The average flow rate from the VESUVIUS analysis is seen to be close to the measured flow rate. The calculated flow rates are in general slightly lower, however the results are considered to be reasonably close. The differences with other codes' results are largely due to friction coefficients, which affect the particle-coolant mixing, and whether or not coolant sub cooling is considered. All of other sub-modules will be completed and verified within the Phase-1 period.

As one of detailed analysis modules, debris spreading model of debris cooling analysis module is also being developed. The debris cooling analysis module has various types of models, such as debris spreading, debris cooling, crust heat conduction, debris bed heat conduction, heat transfer for various types and materials, gap cooling, RPV creep rupture, etc.

The debris spreading, debris cooling, heat transfer for various modes and materials, and gap cooling models were completed. Calculation results of debris spreading model are compared in Fig. 5 with the UC Santa Barbara test results [7]. Water falls on the ruptured center tube of concentric tube geometry, exited through rupture and spread throughout the outer tube. The spreading shape from the debris spreading model analysis is seen to be close to the measured spreading shape. The calculated spreading rates are in general slightly slower in the initial stage, however the results are considered to be reasonably close. All of the other models will be completed and verified within the Phase-1 period.
been initiated. Some of mechanistic models were verified with basic experiments. The basic physics modules for analysis of single- and two-phase flow are incorporated into the Code as sub-modules.

1. Dryout and Critical Power Prediction.

The critical power can be predicted by combining a liquid film flow module, which analyzes dryout phenomena on fuel rod surfaces, and a subchannel analysis module, which analyzes flow and quality distribution in a fuel rod bundle. A spacer effect on liquid film flow is evaluated by combining a three-dimensional steam flow analysis and liquid droplet transport analysis. The subchannel analysis takes account of mixing due to turbulence and pressure difference between subchannels, and void drift. The drift flux model is used for the evaluation of void fraction. The subchannel analysis module was confirmed by analysis of the rod bundle test conducted by R. T. Lahey et al. [9]. For a test with a 3×3 rod bundle, mass flux and thermal equilibrium quality at each subchannel were measured at the bundle outlet position. The analyzed mass flux agreed with the test data to within a standard deviation of 7%. The analyzed quality was within the measured error band of 0.024.

The change of liquid film flow rate along the rod surface is calculated taking account of the entrainment rate of liquid droplets from the liquid film, deposition rate of droplets to the film, and evaporation rate of the film. The Whalley Model is used for the calculation of entrainment and deposition rate [10]. Dryout is defined as the point where the mass flux of the liquid film becomes zero. In order to check the adequacy of this method, a RISO single rod test was analyzed [11]. The calculation results agreed with the test results to within a 7% deviation, as shown in Fig. 6.

C. Boiling Transition Code

The Boiling Transition Analysis Code is being developed for the prediction of dryout and critical power of BWR fuels, and DNB of PWR fuels [8]. The code has a modular structure. The detailed design was completed and coding has
2. DNB Prediction

Void distribution in a hot sub channel of a PWR fuel bundle can be calculated with a two-fluid model. DNB is calculated with the Weisman Model, based on the calculated void distribution [12]. Distribution of mass flux and quality is calculated taking into account the pressure drop in the core, mixing due to turbulence and differential pressure between sub channels, and void drift. Bettis test results, in which a rectangular channel was used, was compared with the calculation for the verification of the two fluid model [13]. In the calculation, bubble detachment was evaluated by the Levy and Saha-Zuber models [14, 15], bubble growth was evaluated by the Lahery model [16], and relative velocity of steam and water was evaluated with the homogeneous model. The calculation results was compared with the test results, which gives good agreement.

The Weisman model for DNB calculation includes empirical parameters; critical void fraction in a bubble layer along a heated surface, Alpha b, and ratio of bubble diameter to thickness of bubble layer, Kappa. The DNB calculation with Chang's recommendation [17], Alpha b = 0.70 and Kappa = 0.15, showed generally good agreement, but slightly higher values than for the test data, as shown in Fig. 7.

V. CONCLUSIONS

Four years of the IMPACT project have been completed with financial sponsorship from the Japanese government's Ministry of International Trade and Industry.

The final result of the IMPACT program will be an integration of software modules for nuclear power plant simulation, capable of analyzing conditions ranging from normal operation to hypothesized accidents. Multiple modules will be executed simultaneously while appearing to the user to be a single code.

Three major sections of the software have been described herein. In the initial technical development phase, work has begun on these sections of the system, namely, the Control System, the Human Interface System, and the Analysis System.

Current results of single-phase flow analyses showed effectiveness of parallel processing. Calculation speedup of at least 30 times by parallel processing with 64 processors was attained for the incompressible single phase flow analysis, depending on a number of mesh. The comparison of the pre-mixing calculation for a steam explosion with the MIXA test results showed good agreement. Calculation results of debris spreading model in debris cooling process are compared and showed good agreement with the UC Santa Barbara test results. The physical models in the Boiling Transition Code has been completed and verified for both BWR fuels critical power and PWR fuels DNB prediction capabilities by comparison with basic experimental results.
REFERENCES


LATEST IMPROVEMENTS ON TRACPWR SIX-EQUATIONS THERMOHYDRAULIC CODE.

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ABSTRACT

The paper presents the latest improvements on TRACPWR aimed at adapting the code to present trends on computer platforms, architectures and training requirements as well as extending the scope of the code itself and its applicability to other technologies different from Westinghouse PWR one.

Firstly major features of TRACPWR as best estimate and real time simulation code are summarized, then the areas where TRACPWR is being improved are presented. These areas comprising:

* **Architecture**: integrating TRACPWR and RELAP5 codes.

* **Code scope enhancement**: modelling the Mid-Loop operation.

* **Code speed-up**: applying parallelization techniques.

* **Code platform downsizing**: porting to Windows NT platform.


* **Code scope extension**: using the code for modelling VVER and PHWR technology.
1. TRAC-PWR MAJOR FEATURES

TRAC-PWR is an advanced version of TRAC "best-estimate" thermalhydraulic code series for simulation of transients in pressurized water reactors, adapted by TECNATOM for being implemented into PWR training and engineering simulators. To accomplish these training and engineering functions, TRAC-PWR combines the major features of engineering best estimate codes along with those of real time simulation codes for training purposes.

1.1 TRAC-PWR major features as "best-estimate code"

Best-estimate engineering code features are implemented into TRAC-PWR, thus providing accuracy and reliability levels comparable to those from other "state of the art" best-estimate thermalhydraulic codes.

These features include, among others:

- Non-homogeneous and non-equilibrium model for the two-phase flow system.
- Boron and non-condensable models.
- Shear and heat transfer at vapor-liquid interfaces and wall surfaces dependent on flow regime.
- Modular structure.
- One-dimensional heat conduction model in fuel rods and structures.
- Three dimensional vessel model.
- Optional point kinetics model or 3-D Kinetics model.
- Counter-current flow limitation model.
- Critical flow model.
- Level tracking model and phase separation in vertical components.
- Implicit numerical scheme.

1.2 TRAC-PWR major features as "real time simulation code"

TRAC-PWR has the capability to simulate all the major events and phenomena needed for PWR simulators and plant analyzers. Besides of this, TRAC-PWR meets the following features which provide the necessary velocity and flexibility for simulation purposes, and make it suitable to be used for full scope training simulators:

- Real time performance throughout the entire calculations.
- Interactive nature.
- Capacity to simulate the full range of operational conditions.
- Time step size stability.
- Realistic instrumentation readings.
- Initial conditions optimized management and conditions saving capacity.
- Execution time management.
- Generic nodalization for the whole range of scenarios to be simulated.
- Communication with additional and simpler models of auxiliary systems beyond the TRAC-PWR model scope.
- Malfunctions definition.

1.3 Six-equations TH codes working in real time; State-of-the-art.

First best estimate six-equations TH codes adapted to work in operators training simulators, were Cathare and TRAC (TRAC-PWR and TRACS for BWR) codes.

These projects took place six years ago in France (EdF) and Spain (Tecnatom). TRACS was a joint development Tecnatom-General Electric.

Since that time Tecnatom has upgraded and built up new simulators with TRAC real time codes.

Recently, other six-equations TH codes have appeared, i.e.: Relap 5 Real time.

2. TRACPWR LATEST IMPROVEMENTS

The latest improvements performed on TRACPWR comprise, amongst others, the following areas:

2.1 Architecture improvements:

- "TRACPWR-RELAP5 INTEGRATION"

Aimed at achieving a higher flexibility in the use of simulators, adding to the training features those derived from the engineering simulators, a project was launched to obtain a multipurpose real time execution architecture, and a "software switch" allowing selection between TRACPWR and RELAP 5/MOD3 codes, and a powerful interactive interface enabling the indistinct use of both codes.

The boundary conditions needed by RELAP are similar to the TRACPWR ones. Replacing the TRACPWR model by an equivalent
RELAP model will permit its use with dynamic boundaries versus the
typical fix boundaries determined for each Relap transient performance.

This improvement, nowadays completed in the Vandellos NPP
simulator, has covered the following objectives:

- RELAP execution within the training simulator architecture. RELAP
  simulation load is not required to run in real time.

- RELAP integration with the rest of simulation models.

- TRACPWR debugger adaptation for use with RELAP.

Each TH code will have its own set of initial conditions, therefore the
simulator will be capable to reset from any of these conditions. No effort
has been applied to implement a possible "on-line" switch of codes, a
quite complex issue and with uncertain benefits.

As a conclusion, this project has permitted the use of RELAP with
dynamic boundaries, thus enhancing the scope of the model, avoiding
also the constraints imposed by fix boundaries conditions.

Furthermore, the Relap integration with the simulator application will
benefit the NSSS model Relap-based with a complete interaction
through the simulator MMI.

2.2 Code scope enhancement:

"MID-LOOP OPERATION".

The Mid-Loop operation concept does not imply only the simulation of
the thermalhydraulics related to this situation but also the dynamic
modes appearing in the process from cold shutdown condition to stable
mid-loop scenario. These physical phenomena appear also during
some accident sequences.

The main objective of this enhancement is to permit the training in the
PWR simulator in thermalhydraulic and operation phenomenology
implied in the mid-loop transients. During these situations, the primary
circuit is in cold shutdown condition with a low value of the decay heat.
Typically the cooling of the Primary System is carried out with the
Residual Heat Removal System and venting by the pressurizer and/or
the vessel head. Venting can be made directly to containment or
pressurizer relief tank.
In both cases there is a high rate of non-condensables in the primary: SG's tubes, pressurizer, vessel head and coolant pipes. The presence of non-condensables gases modifies the system thermalhydraulic performance.

Although the physical features related to mid-loop operation are not so complex than the existing ones in some accident sequences at power, it is necessary to remark that important particular phenomena can be observed in these scenarios.

Basically, main differences could be identified as follows:

- Non-condensables high rate in the primary circuit.
- SG's performance concerning to condensation capacity, etc.
- Long term operation at low pressure (close to atmosphere)
- Stable gas-liquid interface in hot and cold legs of the primary and its influence on the heat and momentum transfer.
- Possible appearance of vortex effect in the RHR pipes.

The main uncertainty at the beginning of development has been the feasibility for resolving in real time the mid-loop operation with the same results fidelity.

Others simulators have chosen a separate solution by developing an independent model for the simulation of this type of scenario. The independent model so developed involves a simpler thermalhydraulics equations taking into account the simulation of this new physical situations.

In the case of Tecnatom's simulators, a hard effort has been made to resolve the mid-loop operation characteristics in the usual numerics calculation of TRAC-PWR with some additional advantages: results fidelity, calculation improvement, smooth transition between different events along a transient, etc.

2.3 Code Speed up:

"PACOTE: TRACPWR CODE PARALLELIZATION PROJECT"

An old technical ambition is to manage the execution of high fidelity simulation models several time faster than real time. It would open the
door to predictive simulations acting as a modelling kernel for Computerized Operator Support Systems.

In the same manner, a TH code with less CPU requirement could be subject to finer, more detailed, nodalization.

Both reasons led Tecnatom to launch the PACOTE project.

The PACOTE project, funded by the PCI (Parallel Computing Initiative; ESPRIT framework), has parallelized the TRACPWR code, increasing the real time slack, thus reducing the probability of real-time loss which might occur in very complex transients.

Two different parallelization approaches were analyzed: tasks or data parallelization.

Because schedule limitation and based on the advantages that PWR plants topology provides, the second one was finally taken. Besides that, the TRAC-PWR numerical resolution scheme solving each loop in an independent manner, favoured that decision.

Most relevant milestones in the project were:

- New component implementation, utilized for primary loops synchronization and balance.

- NSSS parallel model design, being executed in four CPU's. Three CPU's support the calculation of the NSSS secondary loops (steam generator side), while the fourth one calculates the primary loops and the vessel.

- Integration of the TRACPWR parallel model with a real time execution environment and corresponding man-machine interface, forming a Parallel Plant Analyzer (PPA).

The finally selected platform was a Silicon Graphics with for CPU's R4400-200 MHz.

The PPA was validated with diverse ANSI 3.5 transients (Reactor Trip, SBLOCA, ATWS, etc.) with fairly good results in comparison with the TRACPWR sequential performance.

Main conclusions of this enhancement were:

- The best execution time reductions were obtained with a four CPU's platform configuration, where the computational load was balanced.
- With such configuration and a TRACPWR model with 131 cells, the speed-up factor was three times.
- Finer nodalizations improved modestly that factor.

2.4 Code platform downsizing:

"PORTING FROM WORKSTATION PLATFORM TO WINDOWS NT PLATFORM"

This project tried to match current market trends porting the code from UNIX based on platform to Windows NT PC one.

This project was divided in two stages. The first one was aimed to get an executable program capable to work under Windows NT. In the second stage the code was validated with different models of PWR plants and group of transients, against the results obtained by the code with the same models on a UNIX environment, and with actual plant data too.

Finally an efficiency and execution time analysis was made, comparing it with data obtained on the UNIX environment. This analysis demonstrated the TRAC-PWR code porting feasibility.

The Digital Visual Fortran v.5.0 compiler integrated in the Microsoft Developer studio was used.

Needed changes were made on the code for its compilation on Windows NT mainly due to the differences between the Fortran compilers. These were:

- Data management
- Some inner routine format
- Data format accepted for operations, like routines callings or comparisons
- Data handleless storage capacity

The auxiliary programs for graph and extract data were also ported to Windows NT.

The methodology used for the validation was the execution of different models under a UNIX and a Windows NT platforms. Afterwards the results obtained from both platforms were compared.
Some of the validation transients were taken from the ANSI 3.5 standard.

The validation results were right. The differences between both platforms never exceeded the most restrictive criteria for stable situation. These criteria put the limit of difference in 1% or 2% of the reference unit instrument loop range.

Later an execution time analysis was carried out. It was shown that the execution under Windows NT was slower than the corresponding UNIX platform. In three loops models the rate of execution time was over 66% in windows NT and 45% in UNIX.

Most important conclusions of this enhancement were:

- The TRACPWR off-line version works correctly on a PC under Windows NT.

- The execution efficiency is slightly worse on a PC than on a work station.

2.5 On-line performance:

"ON-LINE INITIALIZATION OF A INTERACTIVE GRAPHIC SIMULATOR"

The Vandellós Interactive Graphic Simulator (IGS) will be on-line connected with the Plant Process Computer (PPC) from which a convenient set of data will be read and, once validated, the simulation of the current state of the Plant may start.

Once initialized with actual and current plant data, the IGS will be ready to work as a normal predictive simulator. The use of the IGS may be extended to the on-line testing of calibration of process instrumentation channels.

Therefore the main innovation of this project is the capacity of the simulator to be initialized on-line with actual and current data taken from the PPC.

This initialization will be carried out in a periodic and automatic manner, with a frequency determined by the speed of the computer being used and it could be in the range of a few minutes (e.g., 5 minutes).

A three module structure has been devised:
Input Module, with the following functions:

- management of the initial conditions, from which the simulation will basically start
- communication with the PPC, reading the values of a predetermined subset of the near 1000 analog signals that it manages
- filtering of data read from the PPC, applying validation signal procedures
- preparation of the input data for simulation codes, either in initialization or restart mode

Simulation Module, with the group of simulation codes:

- the thermalhydraulic code selected for the initial phase of this project is TRAC-PWR

Output and Comparison Module, with the following functions:

- management of the simulation results (initial steady state, transients, snapshots, final conditions, etc.)
- get the values of the variables of interest, calculated by the simulation codes, and show them in the appropriate format
- management of the restart files, necessary in order to restart the simulation in certain moments
- comparison of the values of a predetermined group of variables, obtained from the PPC and from the simulation, with a double objective:
  * on one hand, the identification of possible uncalibrations in the process instrumentation channels
  * on the other hand, the identification of deviations in the simulation that suggest the convenience of a readjustment in the output models of the simulator

The hardware architecture will be compatible and imbricated in the same computer platform that will support the off-line Graphic Interactive Simulator of the Plant (currently under construction for Vandellos II NPP) taking advantage of, especially, their access to the PPC.
2.6 Application of the code for modelling VVER and PHWR Nuclear Technology.

The fourth IAEA Standard Problem Exercise (SPE-4) post-test calculation with TRAC PWR was performed to analyze the code capabilities for simulating transients on a VVER reactor like facility.

SPE-4 experiment was carried out at the KFKI Atomic Energy Research Institute PMK-2 facility, in the framework of the IAEA TC Project RER/9/004, and consisted in a 3.2 mm φ Small Break Loss Of Coolant Accident (SBLOCA) in the cold leg, starting from full power and with the unavailability of the high pressure injection system (HPIS). For the prevention of the core damage, the secondary side “bleed and feed” manoeuvre was assumed to be used.

VVER 440/213 reactors are significantly different from PWR’s of western design, and have a number of special features: horizontal steam generators, loop seal in hot and cold legs, safety injection tanks, etc.

The SPE-4 post-test calculation with TRAC-PWR rendered completely successful results, matching closely the recorded experimental data.

The accuracy of the results achieved with TRAC-PWR was at the same level than that of the best calculations presented, its computer efficiency being the one associated to simulation requirements (real time performance).

Therefore, the main objectives of the calculation were achieved, and the TRAC PWR simulation code capability to calculate such a transient on a VVER reactor like facility was proved.

Following with the development of Interactive Graphic Simulators (I.G.S.) based on real time simulation code TRAC-PWR, and in a joint effort between TECNATOM simulation department and ATUCHA I NPP (Argentina) technical staff, a specific I.G.S. for this plant was developed during 1995 and 1996.

Atucha I is a Pressurized Heavy Water Reactor type NPP, Siemens design, natural uranium (0.7 % enrichment) fuelled and 357 MW gross electrical output. Its design includes two reactor coolant loops (heavy water) and two separate moderator cooling loops (heavy water at a lower temperature). Secondary coolant and safety injection systems makeup are light water.

Within the project development, the part related to the application of TRAC-PWR code to a thermalhydraulic system involving heavy water
subsystems (primary coolant and moderator) along with light water based ones (secondary coolant and safety injection) has been the more innovating subject, along with a system physical layout quite different from those of the reference NPPs corresponding to the previously developed I.G.S.

After a thorough analysis, it was decided not to modify the code internally, in order to address the heavy water specific thermodynamic properties, since it would not only imply the substitution of the light water tables and correlations in the corresponding routines for those associated to heavy water, but also to keep the original ones for modelling the light water based subsystems and even to address the mixing of both fluids during certain scenarios (SGTRs, SI actuations, etc.). It is worth to point out that although some modern best estimate thermalhydraulic codes, such as RELAP 5/MOD 3, allows to model systems containing both H₂O and D₂O in separate subsystems, they are not capable to address their mixing.

Consequently, a light water based model development was undertaken.

Regarding the specific physical configuration of the plant systems and components to be modeled with TRAC-PWR, the most characteristic part consisted in the development of the reactor pressure vessel model, by means of the 3-D TRAC-PWR provided VESSEL component, containing five axial levels and two radial segments, associated to three separate CHANNEL components to represent the volumes filled by the primary coolant, and including the upper and lower head "filling structures" (large metallic masses localized in the vessel upper and lower zones). The RPV model was completed with a five cell 1-D PIPE component to represent the "moderator tank". In addition, the modelling of the different mechanisms of heat exchanging in each zone (conduction, convection, moderation and fluid mixing) required a quite complex model adjustment.

The resulting model included 46 1-D components with 183 hydraulic cells, one 3-D component with 10 cells, 16 boundary conditions, 83 junctions and 110 heat structures with 422 thermal nodes. With this nodalization, the following systems were modeled:

- Primary Coolant System (QH)
- Pressurizer and Pressurizer Relief Tank (QD)
- Moderator System (QM)
- Steam Generators (QV)
- Main Steam System (RA)

As it was demonstrated by the validation and acceptance tests results, the project objectives were met, thus showing the viability of TRAC-
PWR use for the development of a heavy water reactor simulator, with the limitations inherent to the applied approach which considers a single type of fluid in liquid or vapor phase. Hence, TRAC-PWR applicability to different NSSS configurations for simulation and training objectives have been demonstrated once more.
EVALUATION OF TWO-FLUID AND DRIFT FLUX THERMOHYDRAULICS IN APROS CODE ENVIRONMENT

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Abstract

The characteristics of the thermohydraulic solutions in APROS are considered for the nuclear power plant modelling. The thermohydraulic model of the APROS plant analyzer includes three levels of solutions, homogeneous 3-equation model, 5-equation drift flux model and 6-equation two-fluid model. In practical modelling of versatile process systems different approaches are selected for different types of the power plant sections. The 3-equation model is used for turbines and auxiliary systems. The 5-equation model and 6-equation model are alternative models for main process sections of the primary and secondary sides. The 5-equation model has been typically selected for the real time applications and the 6-equation model for the safety analysis applications. The validation needs for both approaches are the same. Because the change of the solution mode is an easy task in APROS, the validation tasks are typically performed in parallel for 5-equation and 6-equation models. By calculating in parallel with both models systematic errors in solutions may be pointed out. The testing against both separate effects tests and integral tests is an essential part in the thermohydraulics. In different plant applications different physical features are important. The analysis requirements vary from one application to another. When nodalizations together with increased computer speed are growing up, the earlier validation cases may be insufficient. That is why the content of the code has to be known in detail. Such an expertise in the code development has to be gained that properties of the code against other thermohydraulics codes are known.

1. Introduction

APROS plant analyzer has been developed in a co-operation project between Imatran Voima (IVO) and the Technical Research Centre of Finland (VTT). The simulator has an efficient and user friendly graphical interface and it can be used in the process and automation design, safety analysis and training for the nuclear, fossil and chemical processes. The simulation sections for the nuclear power plant simulation include 1D- and 3D-neutronics, thermohydraulics, automation, electrical network and containment simulation packages. The thermohydraulic model includes three levels of solutions, 3-equation homogeneous model, 5-equation drift flux model and 6-equation two-fluid model.
The development of both thermohydraulic solutions started in the beginning of 1980's as separate projects. The development of the 5-equation model originated from practical need for a fast running thermohydraulic model for scoping studies of small break LOCA accidents. The development of the 6-equation model started from the need to understand the characteristics of the two-fluid solutions at the time, when available RELAP5-versions for small break LOCA studies were under development. The 5-equation model is based on a non-iterative algorithm of five conservation equations with a single momentum equation for the mixture. The phase separation is solved by using the drift flux model. The 6-equation model is based on an iterative algorithm of six conservation equations. The implementation of both solutions into the same platform allows the systematic comparison of approaches with respect to the physical content details, separate effects test cases, integral test cases and plant application results.

In the paper the basic thermohydraulic models of both solutions are presented and compared. The modelling of constitutive equations and numerical solutions is discussed. The code validation program against separate and integral tests is introduced. The plant applications for PWR and BWR plants are presented and the critical physical models, constitutive model set-up and numerical solutions in different transient and accident classes are discussed.

2. Thermal hydraulic model

The basic conservation equations, basic formalisms for wall boundary conditions and for constitutive equations of both models are presented in Table 1. The equations are presented in the same table in order to demonstrate the evolution’s of different models from basic equations. For the 6-equation solution the basic equations are equations (1), (2), (3), (4), (5) and (6). For 5-equation solution they are instead equations (1), (2), (7), (5) and (6). For phase separation the drift flux model applies the formalism presented in the equation (8). The phase separation in the 6-equation model is based on the balance between wall friction to liquid, wall friction to gas and interfacial friction, i.e. the equations (9) and (10). In 5-equation model the friction is needed only between the wall and mixture and thus only equation (11) is needed. The basic formalism for the wall heat flux is according to the equations (12) and for the interfacial heat flux according to equations (13). The mass transfer due to interfacial heat transfer is defined as an energy balance over the interface. The wall heat flux partitioning to boiling and liquid heat-up is a function of wall heat flux, liquid subcooling, flow rate and channel diameter.

The evaluation of the models indicates that on the basic equation level quite little differences exist between different solutions. Most important is the difference of describing the phase separation by using interfacial friction or drift flux correlation. The physical reality in the basic equations is defined by the constitutive models. The selection of constitutive models is presented in Table 2. Due to different development histories there are more differences in constitutive models than would be necessary. The formalism presumes that in 6-equation model the wall friction is divided into two fractions, one is the friction for liquid and another the friction for gas. The interfacial friction model is very essential, because it defines phase separation.
Mass conservation equations

\[ \frac{\partial (\Lambda \rho)}{\partial t} + \frac{\partial (\Lambda \rho u)}{\partial z} = A_y + A_z \]  \hspace{2cm} (1)

\[ \frac{\partial (A(1-\alpha)\rho)}{\partial t} + \frac{\partial (A(1-\alpha)\rho u)}{\partial z} = -A_y + A_z \]  \hspace{2cm} (2)

Momentum conservation equations, two-fluid model

\[ \frac{\partial (\Lambda \rho u)}{\partial t} + \frac{\partial (\Lambda \rho u^2)}{\partial z} = -A\alpha \frac{\partial p}{\partial z} + Aq_x + Aq_z + A\gamma h \frac{\partial u}{\partial x} + A_5 h \frac{\partial h}{\partial z} \]  \hspace{2cm} (3)

\[ \frac{\partial (A(1-\alpha)\rho u)}{\partial t} + \frac{\partial (A(1-\alpha)\rho u^2)}{\partial z} = -A(1-\alpha) \frac{\partial p}{\partial z} + Aq_x + Aq_z - A\gamma h \frac{\partial u}{\partial x} + A_5 h \frac{\partial h}{\partial z} \]  \hspace{2cm} (4)

Energy conservation equations

\[ \frac{\partial (\Lambda \rho h)}{\partial t} + \frac{\partial (\Lambda \rho h^2)}{\partial z} = -A\alpha \frac{\partial h}{\partial z} + Aq_x + Aq_z + A\gamma h \frac{\partial h}{\partial x} + A_5 h \frac{\partial h}{\partial z} \]  \hspace{2cm} (5)

\[ \frac{\partial (A(1-\alpha)\rho h)}{\partial t} + \frac{\partial (A(1-\alpha)\rho h^2)}{\partial z} = -A(1-\alpha) \frac{\partial h}{\partial z} + Aq_x + Aq_z - A\gamma h \frac{\partial h}{\partial x} + A_5 h \frac{\partial h}{\partial z} \]  \hspace{2cm} (6)

Momentum conservation equation, drift flux model

\[ \frac{\partial (\Lambda \rho u)}{\partial t} + \frac{\partial (\Lambda \rho u^2)}{\partial z} = -A \frac{\partial p}{\partial z} - \rho \frac{\partial u}{\partial z} - \rho \cos \theta \]  \hspace{2cm} (7)

Drift flux model for phase separation

\[ u = C_1 j + v \]  \hspace{2cm} (8)

Wall friction for both phases \((k=1,2)\), two fluid model

\[ \frac{\partial p}{\partial z} = \frac{1}{2} C_2 \frac{\partial u}{\partial z} \left| u \right| \quad \text{with} \quad C_n = C_n \left( R e_n \right) \]  \hspace{2cm} (9)

Interfacial friction, two fluid model

\[ \frac{\partial p}{\partial z} = \frac{1}{2} C_2 \alpha (1-\alpha) \rho \left( u - u \right) \left| u \right| \]  \hspace{2cm} (10)

Wall friction, drift flux model

\[ \frac{\partial p}{\partial z} = \frac{1}{2} \left( \frac{f}{D} + X \right) (\alpha \rho u + (1-\alpha) \rho u) \]  \hspace{2cm} (11)

Wall heat flux to gas \( \text{Wall heat flux to liquid} \) \( \text{Wall heat flux to boiling} \)

\[ q_a = h_u \left( T - T_i \right) \quad q = h_a \left( T - T_i \right) \quad q = h_a \left( T - T_i \right) \]  \hspace{2cm} (12)

Interface to liquid heat flux \( \text{Interface to gas heat flux} \)

\[ q_a = h \left( T - T_i \right) \quad q = h \left( T - T_i \right) \]  \hspace{2cm} (13)

Boiling along the interface

\[ q = q_a + q \]  \hspace{2cm} (14)

Table 1. Conservation equations, friction and phase separation models

For the heat transfer different correlation's have been used, but for typical light water reactor cases the differences in the wall heat flux calculation are quite small. The interfacial heat transfer models differ much more and the reason is that the 5-equation model was derived as a non-iterative solution, which should tolerate large timesteps. The condensation correlation gives a reasonable condensation rate and the condensation is not strongly flow rate dependent. For 6-equation model the condensation is strongly dependent on the flow rate.

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In general developing of constitutive models is an extensive task. Different types of help tools have been created for making the work easier. The correlation's may be mapped over their parameter variation ranges. For interfacial heat transfer and interfacial friction correlation's the numerical values may be compared against RELAPS-results.

<table>
<thead>
<tr>
<th>Physical phenomena</th>
<th>5-equation model</th>
<th>6-equation model</th>
</tr>
</thead>
<tbody>
<tr>
<td>Wall friction</td>
<td>Blasius equation for mixture</td>
<td>Blasius eq. for both phases, factorings: void fraction and flow regime</td>
</tr>
<tr>
<td>Interfacial friction</td>
<td>Not modelled for the drift flux model</td>
<td>Separately: bubbly, annular, droplet, stratified, weighting function</td>
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<tr>
<td>Net vapourisation</td>
<td>A linearised ramp function from subcooled liquid to saturation point</td>
<td>Separate gas interface, liquid interface, energy balance for interface</td>
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<tr>
<td>Pre-DNB heat transfer</td>
<td>Dittus-Boelter, Chen as simplified for boiling</td>
<td>Dittus-Boelter, Thom</td>
</tr>
<tr>
<td>Critical heat flux for wall heat transfer</td>
<td>Zuber-Griffith, VVER: Smolin, Bezrukow</td>
<td>Zuber-Griffith, Biasi, VVER: Smolin, Bezrukow</td>
</tr>
<tr>
<td>Post-DNB wall heat transfer</td>
<td>Dittus-Boelter to gas</td>
<td>Dittus-Boelter, Berenson</td>
</tr>
<tr>
<td>Special heat transfer models</td>
<td>No quench model yet</td>
<td>Heat transfer enhancement near quenching front</td>
</tr>
<tr>
<td>Interfacial condensation</td>
<td>Droplet type condensation or through stratified water level</td>
<td>Shah mode for liquid, Lee-Ryley model for gas</td>
</tr>
<tr>
<td>Interfacial flashing</td>
<td>Linear function of liquid mass and liquid superheat</td>
<td>Exponent function of void</td>
</tr>
<tr>
<td>Critical flow limitation</td>
<td>Sound velocity limitation or Moody model applied for the junction</td>
<td>With dense nodalization inherently or limitation to sound velocity</td>
</tr>
<tr>
<td>Pump characteristics</td>
<td>Four quadrant curves for head and torque for flow and pump speed.</td>
<td>Four quadrant curves for head and torque for flow and pump speed.</td>
</tr>
<tr>
<td>Phase separation</td>
<td>Drift flux model derived from EPRI correlation or full separation</td>
<td>Via interfacial friction correlation or full separation</td>
</tr>
<tr>
<td>Material property solution</td>
<td>Rational function fittings, two- or one parameter functions</td>
<td>Table interpolation as a function of pressure and enthalpy</td>
</tr>
</tbody>
</table>

Table 2. Constitutive thermohydraulic models for 5- and 6-equation solutions

There are some areas, where the creation of a constitutive model has been found specially problematic. One reason is that the available correlation's in the literature are typically valid on a limited parameter range and the validity range is not clearly expressed. However the selected correlation's should be valid over the whole parameter range, from condenser pressure up to supercritical pressure, from ice temperature to the fuel melting temperature, from zero flow to sound velocity.

The problematic areas may be listed as follows:
- Applying sound velocity for choking flow of the flashing two-phase mixture.
- Calculation of net evaporation shape function for the wall heat flux.
- Deriving rational function fittings to material correlation's.
- Pump head and torque characteristics for two phase flow.
- Quench front model taking into account the axial conduction.

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The numerical solution for the 5-equation model is a predictor-corrector type noniterative solution. The solution for the 6-equation model is iterative. In the 5-equation model the sparse matrix inversion is used for solving the pressure, void fraction and energy equations. In the 6-equation model the sparse matrix inversions are also used for the pressure, void fraction and energy equations. In Table 3 the main features related to the 5- and 6-equation numerical solution are presented.

<table>
<thead>
<tr>
<th>Numerical feature</th>
<th>5-equation model</th>
<th>6-equation model</th>
</tr>
</thead>
<tbody>
<tr>
<td>Numerical solution principle</td>
<td>Predictor-corrector solution without iteration</td>
<td>Iteration until the final convergence</td>
</tr>
</tbody>
</table>
| Solution scheme | 1. Predict heat transfer and mass transfer  
2. Solve pressures  
3. Update heat and mass transfer  
4. Calculate mixture velocity  
5. Calculate drift flux separation  
6. Solve mass flows  
7. Integral mass conservation  
8. Solve phase enthalpies  
9. Update material properties  
10. Proceed to the next timestep | 1. Solve pressures  
2. Calculate phase velocities  
3. Update linear densities  
4. Solve void fractions  
5. Solve heat structures  
6. Calculate heat flows  
7. Solve phase enthalpies  
8. Update material properties  
9. Check convergence, if not, return to beginning |
| Matrix inversions | Pressure  
Void fraction  
Heat structure temperature  
Gas enthalpy  
Liquid enthalpy | Pressure  
Void fraction  
Heat structure temperature  
Gas enthalpy  
Liquid enthalpy |
| Discretized scheme | Staggered grid for space discretization  
Implicit in time advancement | Staggered grid for space discretization  
Implicit in time advancement |
| Timestep in the calculation | 0.01 ... 0.2 s, fixed time step, short timesteps in rapid transients | 0.01 ... 0.1, automatic time step control |
| Source term linearizations | Linearizations in respect pressure, liquid temperature, gas temperature and wall temperature. | Linearizations in respect pressure, void fractions, velocities, surface temperatures |

Table 3. Numerical solutions

Although one model is non-iterative and another iterative, both solutions include similar type of solution steps as indicated in Table 3. The implicit integration principle, i.e. the new values (superscript n+1) are applied for the solved variables, is shown in Table 4. The staggered grid means that state variables are defined in node centres and flow variables in junctions connecting nodes. For nonlinear type of equations and constitutive models all the parameters dependencies can not be considered implicitly. For strong dependencies the source term linearization is a practical approach. The main principle for the source term linearization is shown in Table 4.

The argument x used in the source term linearization means pressure, phase enthalpy or wall surface temperature. The linearization is applied in order to achieve a stabilized solution by long timesteps or event between different iteration steps.
Space and time discretization
\[
\frac{\phi^*_n}{\Delta t} = \frac{\phi^*_{n+1} - \phi^*_n}{\Delta t} - \frac{\phi^*_n - \phi^*_{n-1}}{\Delta t} \tag{14}
\]

Source term linearization is used \[
\phi^{***} = \phi^* + \left(\frac{\partial \phi^*}{\partial x}\right)^* (x^{**} - x^*) \tag{15}
\]

Table 4. Basic formulas in the numerical solution

In the numerical solutions the problems concern mostly the convergence of iterations, interface between different types of thermohydraulic solutions and interconnections of thermohydraulics to the neutronics, process component (fast acting valves) and automation.

Together with the computer development there is a trend towards bigger nodalizations, application of new computers (HP, DEC, NT) and new FORTRAN compilers. Especially in the thermohydraulic parts these trends may cause problems. The higher optimisation levels and attempt to parallelize the code may fail. For avoiding potential errors in the solution the program content has to be known very detailed from inside. Usually the computer / compiler related error is referred into a very small detail of the code. When this kind of critical locations are localised, by reprogramming the transportability of the solution may be increased tremendously.

3. Separate effects tests

The separate effects tests are focused on specific physical phenomena. Their advantages are detailed measurements around the studied phenomena with an accuracy, not possible in integral experiments. The model development needs typically a lot of measurement data and that is why the separate effect tests are popular for the code development. When only limited phenomena are studied, the boundary conditions are clearly defined and the experiment may be modelled accurately.

Two types of separate effects test may be mentioned. Some separate effects tests are focused on the specific physical phenomenon like heat transfer or critical flow. Another class of separate effects tests includes experiments for a special process component like the loop seals, or the counter-current-flow-limitation at core exit. Some physical submodels are so important that separate effects testing is required for the model validation.

The Edward's pipe test case was a horizontal pipe which was closed in the beginning of the test and contained water under high pressure. The valve at the pipe end was rapidly opened and liquid was released out by critical speed due to flashing. The critical properties in the simulation are the continuation term in the momentum equation and modelling of the flashing in the pipe.

The purpose of the Battelle blowdown experiment was to examine the steam blowdown from the top of the pressure vessel. The vessel is partially filled by hot water and after pressure drop the water content is swelling. When the swell level reaches the discharge pipe, the discharged flow is changed from steam to two-phase mixture.
The aim of Marviken critical flow tests was to obtain the data for critical flows in large diameter pipes. The test section consisted of a 24 m high pressure vessel, which was partially filled by hot water. The discharge pipe was located at the bottom.

In Becker's experiment the heat transfer crisis and post-dryout heat transfer was studied in a single tube. The experiment makes it possible to compare the whole boiling curve from convection to liquid to convection to superheated steam and partitioning of the wall heat flux between steam heat-up and boiling.

![Graph](image)

Figure 1: Residual water and pressure difference over the loop seal in the IVO experiment. The experimental pressure drop was strongly oscillating between minimum and maximum values.

The IVO loop seal experiments are related to the PWR plant specific detail with loop seals in the intermediate leg after the steam generator. The water remaining in the loop seal in two-phase conditions may lead to the reactor core uncovering. The loop seal phenomenon has been studied in the small scale experiments but there were doubts that the small scale results are difficult to be extrapolated into the real plant scale. The IVO loop seal experiments were conducted for the real loop seal geometry of the VVER-1000 type of PWR at atmospheric pressure by injecting compressed air. The tube diameter was 0.8 m. The main parameters measured in the test were the pressure difference over the loop seal and residual water remaining in the tube. The results for the remaining water mass in the loop and pressure difference over the test section are displayed in Figure 1. The result for the pressure difference shows clearly the difficulties in analysing real experiments. The measured pressure difference was strongly oscillating due to flow pattern changes from stratified flow to mixed flow in the horizontal section and due to U-type of oscillation of the water content. The calculated result was oscillating also, but the amplitudes cannot be matched. In general the oscillatory thermohydraulic phenomena are difficult to calculate, because the numerical solution tends to stabilise excess oscillations and transitions between different flow patterns are impossible to be matched.

The OECD/NEA-CSNI Separate Effects Test Matrix for Thermal Hydraulic Code Validation /1/ defines the best separate effect tests for studying different physical phenomena. The matrix includes typically 3 - 5 recommended separate effects tests for the code validation. In Table 5 matching of the APROS validation cases belonging to different categories are presented. All

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<table>
<thead>
<tr>
<th>Phenomena to be analysed</th>
<th>APROS validation cases matching the group</th>
<th>5-eq.</th>
<th>6-eq.</th>
</tr>
</thead>
<tbody>
<tr>
<td>Basic phenomena</td>
<td>Edward’s pipe, Battelle</td>
<td>x</td>
<td>x</td>
</tr>
<tr>
<td>- evaporation</td>
<td>Edward’s pipe</td>
<td>x</td>
<td>x</td>
</tr>
<tr>
<td>- flashing</td>
<td>Lotus air/water annular flow</td>
<td>x</td>
<td></td>
</tr>
<tr>
<td>- condensation</td>
<td>Edward’s pipe, Battelle, ISP-6</td>
<td>x</td>
<td></td>
</tr>
<tr>
<td>- friction</td>
<td>Lotus air/water annular flow</td>
<td>x</td>
<td></td>
</tr>
<tr>
<td>- pressure drop</td>
<td>Edward’s pipe, Battelle, ISP-27</td>
<td>x</td>
<td></td>
</tr>
<tr>
<td>Critical flow</td>
<td>Edward’s pipe, Marviken, ISP-27</td>
<td>x</td>
<td></td>
</tr>
<tr>
<td>Vertical phase separation</td>
<td>Battelle, ISP-6</td>
<td></td>
<td>x</td>
</tr>
<tr>
<td>Horizontal stratified flow</td>
<td>BETHSY ISP-38</td>
<td>x</td>
<td>x</td>
</tr>
<tr>
<td>Phase separation at branches</td>
<td>BETHSY, ISP-38</td>
<td>x</td>
<td>x</td>
</tr>
<tr>
<td>Entrainment/de-entrainment</td>
<td>Lotus air/water annular flow</td>
<td>x</td>
<td>x</td>
</tr>
<tr>
<td>Liquid-vapour mixing, condensation</td>
<td>NOKO EC bundle</td>
<td>x</td>
<td>x</td>
</tr>
<tr>
<td>Stratified condensation</td>
<td>Loviisa pressurizer</td>
<td></td>
<td>x</td>
</tr>
<tr>
<td>Spray effects</td>
<td>IVO CCFL</td>
<td>x</td>
<td>x</td>
</tr>
<tr>
<td>Countercurrent flow</td>
<td></td>
<td></td>
<td>x</td>
</tr>
<tr>
<td>Multidimensional void/temf.</td>
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<td>x</td>
</tr>
<tr>
<td>Heat transfer:</td>
<td></td>
<td></td>
<td>x</td>
</tr>
<tr>
<td>- pre DNB</td>
<td>FRIGG, Christensen</td>
<td></td>
<td>x</td>
</tr>
<tr>
<td>- critical heat flux</td>
<td>Becker</td>
<td></td>
<td>x</td>
</tr>
<tr>
<td>- post DNB</td>
<td>Becker</td>
<td></td>
<td>x</td>
</tr>
<tr>
<td>Quench front propagation</td>
<td>ERSEC - 7, REWET-II</td>
<td>x</td>
<td>x</td>
</tr>
<tr>
<td>Lower plenum flashing</td>
<td></td>
<td>x</td>
<td>x</td>
</tr>
<tr>
<td>Guide tube flashing</td>
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<td>x</td>
<td>x</td>
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<tr>
<td>Pump one-and two-phase char.</td>
<td>Loviisa pump trip, Left</td>
<td>x</td>
<td>x</td>
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<td>Separator and dryer behaviour</td>
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<td>x</td>
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<tr>
<td>Accumulator behaviour</td>
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<td></td>
<td>x</td>
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<tr>
<td>Loop seal filling and clearance</td>
<td>IVO loop seal tests, Pactel</td>
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<td>x</td>
</tr>
<tr>
<td>ECC bypass / downcomer penetrations</td>
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<td></td>
<td>x</td>
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<tr>
<td>Parallel channel instability</td>
<td></td>
<td></td>
<td>x</td>
</tr>
<tr>
<td>Boron mixing and transport</td>
<td></td>
<td></td>
<td>x</td>
</tr>
<tr>
<td>Lower plenum entrainment</td>
<td></td>
<td></td>
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<tr>
<td>Noncondensable gas effects</td>
<td></td>
<td></td>
<td>x</td>
</tr>
</tbody>
</table>

Table 5. Requirements for the core validation

Experiments used for the APROS validation do not belong to recommended test cases and quite many test cases should be validated further. But on the other hand the experience from APROS validation is, that it is more important to have some validation cases from each group than to perform the validation completely for all recommended cases.

4. Validation against BETHSY integral tests

The integral test experiments in the BETHSY facility for the 2 inch cold leg break (ISP-27) in a PWR and midloop operation test (ISP-38) have been calculated with both 5-equation and 6-equation models. ISP-38 was organised by the OECD/NEA/CSNI/PWG2. It was an open exercise based on test 6.9c on the French BETHSY test loop. The test belongs to a program for studying the PWR response during a loss of the Residual Heat Removal System (RHRS). It
includes the physical phenomena occurring at low pressure and power and a test if the gravity feed draining or forced feed injection are capable to prevent core boiling.

Figure 2. Main steps in the BETHSY ISP-38 test

The transient had three distinct phases based on the overall behaviour of the primary system, Figure 2. During the initial phase the two phase level in the pressure vessel is located close to the axis of hot legs. The two phase level falls below the hot leg nozzles during the second phase, and the core heat-up is initiated after the mixture level reaches the top of core. The third phase begins when the gravity feed injection is started. The injection was sufficient to rapidly reflood the core and refill the whole pressure vessel up to the mid loop condition. The modelling problems were related to the stratification in the hot leg, CCFL in SG inlet pipe, level swell in the core.

The calculation results for 5-equation and 6-equation models of APROS, for RELAP5/Mod3 and experiment are compared in Figure 3 for the void fraction in the hot leg and total coolant mass in the primary loop. The calculation demonstrated, how different problems are met, when the phenomena are analysed outside typical transient and accident conditions. In these analyses the essential features were low pressure, horizontal stratification in hot legs, flow separation in
the T-branch into the pressurizer, phase separation in the core and counter-current-flow limitations in the rising leg before steam generator and pressurizer surge line. The model improvement will be continued for better matching the experimental data especially for the 5-equation model.

![Graph showing void fractions and total system mass in the BETHSY ISP 38 test.](image)

Figure 3. Hot leg void fractions and total system mass in the BETHSY ISP 38 test.

5. Requirements to thermohydraulic models in different plant incidents

In Table 6 the plant specific features for different transients are collected for the plant simulation. Before incidents are calculated, the analyst should be aware if the thermohydraulic model is good enough taking into account specific thermohydraulic features in the incidents. Some important phenomena may be validated by using separate effects tests. On the other hand there are many phenomena, which need validation by integral tests. The natural circulation modes with single phase circulation, two-phase circulation and reflux condensing model are examples from this kind of phenomena.

6. Plant applications for VVER type of PWR plants

The APROS 5-equation model has been mainly used in simulator applications. The 6-equation models have been used as a nuclear plant analyser and for safety analyses. The models for the analysis application and training application cannot be distinguished any more. The detailed models for auxiliary systems, automation and electrical systems are needed in both cases. The simulator model may be used for analysis applications with comparison against system codes by changing the 5-equation solution with that of 6-equation. The modification requires only a few hours. Vice versa the simulator model may be generated by changing 6-equation solution to 5-equation solution.
<table>
<thead>
<tr>
<th>Incident</th>
<th>Simulation requirements</th>
</tr>
</thead>
<tbody>
<tr>
<td>50 - 200 % PWR LOCA blowdown</td>
<td>Fast depressurization, pressure wave propagation, delayed flashing, critical heat flux in the core, two-dimensional downcomer flow.</td>
</tr>
<tr>
<td>50 - 200 % PWR LOCA emergency cooling</td>
<td>Quench front propagation, droplet entrainment, energy non-equilibrium in post dryout, radiation heat transfer</td>
</tr>
<tr>
<td>0.1 - 10 % PWR LOCA (small break LOCA)</td>
<td>Natural circulation modes (single phase, two-phase, reflux condensing mode), system depressurization</td>
</tr>
<tr>
<td>Steam line break (PWR)</td>
<td>Primary side overcooling, core recriticality, pressure vessel cold stress</td>
</tr>
<tr>
<td>Reactor scram, turbine trip, main coolant pump trip (PWR)</td>
<td>Heat transfer crisis, actuation of different signals, hot channel and hot rod behaviour</td>
</tr>
<tr>
<td>Control rod ejection, control rod withdrawal (PWR)</td>
<td>Neutronics 3-D -behaviour due to reactivity changes.</td>
</tr>
<tr>
<td>Loss of off-site power</td>
<td>Decay heat removal of the reactor core</td>
</tr>
<tr>
<td>ATWS after loss-of feedwater (PWR)</td>
<td>Neutronics and thermohydraulics interactions, axial shape of neutron flux, natural circulation modes</td>
</tr>
<tr>
<td>Reactivity accident by boron dilution or cold water plug (PWR)</td>
<td>Non-diffuse numerics for calculating the propagation of the boric acid and cold water front</td>
</tr>
<tr>
<td>Steam line or feedwater line LOCA</td>
<td>Starting the high pressure injection, core cooling by spray or bottom flooding</td>
</tr>
<tr>
<td>Operational transient as recirculation pump trip (BWR)</td>
<td>Heat transfer crisis in the core, dynamics of the reactor scram</td>
</tr>
<tr>
<td>Steam line isolation (BWR)</td>
<td>Power spiking (may be 200-400 % peaking power, before feedback from the void effects) in connection with pressure spiking and core void collapse, pressure wave propagation in the steam line, steam relief capacity, direct heat absorption into coolant, effect of wet steam on pressure behaviour</td>
</tr>
<tr>
<td>Loss of off-site power (BWR)</td>
<td>Securing emergency feedwater injection. After core heat-up melting of control rods and possibility for the criticality.</td>
</tr>
</tbody>
</table>

Table 6. Simulation requirements in different type of accidents

VVER -related applications for APROS have included /2/
- Large Break LOCA cases, comparisons against DRUFAN and FLUT analyses
- REWET tests for reflooding
- Small break LOCA, comparisons against RELAP -analyses
- Primary to secondary leak analyses
- Real plant incident including erroneous opening of the turbine bypass valve
- Main coolant pump trip analyses
- Turbine trip analyses
- Natural circulation calculation, comparison with RELAP analyses
- ATWS LOOP, comparison against HEXTRAN -SMABRE analyses.

In the VVER type of PWR calculations the prediction of different natural circulation modes (single phase circulation, two-phase circulation and reflux-boiling condensing mode) may be emphasised as important in all incidents, where two-phase conditions exist in the primary loop. This characteristics has been studied extensively in PACTEL experiments.
7. Plant applications for ASEA type BWR plants

A transient analysis model of the Olkiluoto BWR has been created with APROS. The model was initially developed for calculation of three verification cases, but it is designed to be easily extended. The main systems: reactor vessel, steam lines, feed water lines and relief system have been modelled using both the 5- and 6-equation models. Auxiliary systems have been described with the homogeneous model. The primary system model starts from the feed water pumps and ends at the high pressure turbine and steam dump control valves.

Three transients have been analysed with the Olkiluoto model. These are:
- double ended steam line break with simultaneous loss of electricity
- loss of feed water, ATWS
- simultaneous closure of all steam line isolation valves

8. Conclusions

The parallel development of the five- and six-equation thermohydraulics solution has proven that accurate simulation results may be achieved by both models. The same development platform for both models makes the model comparison efficient, because practically the same input models can be used for both solutions. The five-equation model is best for real time simulator applications due to fast calculation. The six-equation model allows an application of same physical models as used in internationally available simulation codes and is the most reliable tool for detailed plant analyses.

The present day analysis needs are developing fast and the models being satisfactory several years ago may not be satisfactory at present. New computers bring new possibilities for more detailed models. But on the other hand the new development brings risks into the numerical solutions. Content of simulation models has to be known carefully. In APROS plant analyser the function of thermohydraulics may be studied on the level of system functions and on the detailed level. The interconnections between thermohydraulics and neutronics, thermohydraulics and controllers may be easily displayed. The differences between models may be easily analysed as well. The numerical functions may be analysed carefully as well.

References

/ 2/ Experiences of APROS in Nuclear Power Plant Safety Analysis, H.Kantee, S.Savolainen, H.Kontio

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A Nodalization Study of Steam Separator in Real Time Simulation


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Columbia, MD 21045, USA

Abstract

The motive of this paper is to investigate the influence of steam separator nodalization on reactor thermohydraulics in terms of stability and level response. Three different nodalizations of steam separator are studied by using THEATRe and REMARK Code in a BWR simulator. The first nodalization is the traditional one with two nodes for steam separator. In this nodalization, the steam separation is modeled in the outer node, i.e., upper downcomer. Separated steam enters the steam dome node and the liquid goes to the feedwater node. The second nodalization is similar to the first one with the steam separation modeled in the inner node. There is one additional junction connecting steam dome node and the inner node. The liquid fallback junction connects the inner node and feedwater node. The third nodalization is a combination of the former two with an integrated node for steam separator. Boundary conditions in this study are provided by a simplified feedwater and main steam driver. For comparison purpose, three tests including full power steady state initialization, recirculation pumps runback and reactor scram are conducted. Major parameters such as reactor pressure, reactor level, void fractions, neutronic power and junction flows are recorded for analysis.

Test results clearly show that the first nodalization is stable for steady state initialization. However it has too responsive level performance in core flow reduction transients. The second nodalization is the closest representation of real plant structure, but not the performance. Test results show that an instability occurs in the separator region for both steady state initialization and transients. This instability is caused by an unbalanced momentum in the dual loop configuration. The magnitude of the oscillation reduces as the power decreases. No superiority to the other nodalizations is shown in the test results. The third nodalization shows both stability and responsiveness in the tests.

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1. Introduction

Steam separator is a passive device where a steam mixture is separated into liquid and high quality steam. Steam separation takes place through the spiral path within the device and high quality steam enters steam dome. The separated liquid falls back to the upper downcomer. How to model this process correctly is essential to simulator performance. It is because the indicated level transmitters are connected to the steam separator region. As the thermohydraulic model evolves from single node to loop and then to nodal momentum approach, the separator nodalization becomes more important. The configuration affects not only the level response but also the stability.

In old single node model, this process can be easily addressed by a mass balance (core flow minus steam flow is downcomer flow) equation. Pure steam separation is assumed and momentum effects are not considered. As thermohydraulic model improves to the loop momentum approach, steam separator is modeled in more detail by two different nodes - one for inner separator region and one for outer region (upper downcomer). The steam dome is connected to this outer node as an appendix node. The steam separation is modeled in this outer node. Since both nodes belong to the same loop, it doesn’t matter the inner node is linked to the steam dome node. The two phase mixture is separated into pure steam and pure liquid at the outer node. In reality the inner node is the region where these steam separation devices are located. This arrangement is lack of physical meaning compared to real plant structure yet it produces reasonably well normal and transient results.

As thermohydraulic model advanced to nodal momentum approach in the 90’s, the drawbacks of this nodalization are exposed. First of all, the level transmitter legs are connected in the upper downcomer wall. Any disturbances in the inventory of the outer node could affect the level indication. The hydraulics model could produce unrealistic flows between the inner node and outer node or outer node and the vessel dome in some transients. These flows perturb the liquid inventory in the separator node. Since the steam separation model depends on the liquid inventory of the node, it magnifies the perturbation. The outcome is the indicated level become too responsive for controller. Secondly, the two phase mixture has to travel farther and alter flow direction to be separated in the upper downcomer region. The results are not physically meaningful for training.

To remedy the above mentioned drawbacks, one attempt is to model the steam separation in the inner node. The carry under flow is connected to the feedwater node and a junction between inner node and steam dome node is created for steam separation. This arrangement could help the excessive inventory change in the outer node because only two flow junctions are connected to this node. Another benefit of this arrangement is this nodalization is the closest realization of the steam separator structure. It provides the dynamics inside and outside steam separator. However, this
configuration creates two conjugated loops inside reactor vessel. The existence of a balanced mass, energy and momentum solution remains to be investigated.

Another approach to improve the traditional nodalization is merging the inner separator node and outer separator node as one node. By this arrangement, the momentum balance between these two nodes is automatically conserved and a two-loop dynamics within the vessel is avoided. A more stable level response is expected. But one question arises. Is the level responsive enough in fast transients?

2. Nodalization

Figure 1, 2 and 3 are the test nodalizations. They are used to model an ABB BWR 3000 reactor. These nodalizations are the same except steam separator region. Node 1 is the steam dome that has four steam lines connected to it. Feedwater lines connect to Node 3. Eight internal recirculation pumps are distributed circumferentially in Nodes 5 to 8 with two pumps lumped in one hydraulic node. Node 9 is the bottom head and Node 10 is the lower plenum. One bypass channel node and five power zones are arranged in the reactor core. Among them, Node 12 to 14 represent central power zone and Nodes 15 to 26 are the four 90-degree-quadrant active core channels. This arrangement is designed for strong asymmetric thermohydraulic behaviors in the reactor vessel. It is caused by the four quadrant feedwater spargers in downcomer and four auxiliary feedwater lines connecting the core top in different quadrants. This asymmetric phenomenon is not studied in this paper since no asymmetric cooling processes are involved. Node 27 represents the upper plenum and Node 28 for the standpipes.

In separator region, Nodalization 1 is the traditional configuration. The separator region is represented as Node 29 and it connects Node 2 with a horizontal Junction 25. There is no connection between separator node and the steam dome. The separator model is implemented in Node 2 where the upper downcomer is. The liquid fallback junction is 27 and steam junction is 26. In Nodalization 2, the separator model is implemented in Node 29 with liquid fallback junction 25 and steam Junction 26. In this arrangement, a second loop is created, i.e. Node 1-2-3-29. Nodalization 3 is a variation of Nodalization 2. Node 2 and Node 29 are combined as Node 2 in which the separator model is implemented. In this configuration, the connection between separator inside region and outside region is removed as in the real plant structure and only one main loop exists is in the vessel.
Figure 1 THEATRe Nodalization 1

Figure 2 THEATRe Nodalization 2

Figure 3 THEATRe Nodalization 3
3. Physics model

The separator model in THEATRe is a non mechanical model. Steam junction donor void fraction and the liquid junction donor liquid void fraction are simple mathematical functions as shown in Figure 4. The donor phasic void fraction of the outlet junctions depends on the inventory of separator node only. As the nodal void fraction is greater than voover, the junction void fraction is specified as 1.0. On the liquid fallback junction, the donor liquid void fraction is specified as 1.0 when the nodal liquid void fraction is greater than vunder. This model is independent of local geometry and power conditions. vunder and voover are specified as 0.65 and 0.15 in this study. These two constants are determined based on previous trainer experience.

A collapsed reactor level, i.e. sum of liquid void fraction multiplied by nodal length, is used in this study. Node 1 to 4 are included in the level calculation. No steam flow and recirculation flow effects are considered in the level calculation. The collapsed level shows pure thermohydraulic performance without non related factors. Top of the core (upper elevation of Node 11) is the zero reference line.

The feedwater driver is a simplified variable speed feed pump and a simple PI level controller. In the level controller, normal level set point is 4.2 m and no feed-steam mismatch compensation is modeled. Feedwater temperature is input from the plant data for both steady state and transients. In reactor scram test, the feedwater system enters a scram mode and the feed flow follows a prescribed curve shown in Figure 5. The scram mode deactivates when the reactor level recovers to 3.8 m and wait for 25 seconds.

The steam flow is calculated by the square root of pressure difference between reactor dome and turbine header pressure. There is no steam flow measurement in this plant. Only header pressure data is available. 65.3 bar is set for full power steady state initialization. In recirculation pumps run back and reactor scram case, the header pressure is input as the plant condition for boundary condition. No pressure control is modeled in the driver.

The thermohydraulic model (THEATRe) in this experiment is a replica of RELAP5 model. The only difference is in the momentum equation. THEATRe uses mixture momentum equation plus drift flux equations and RELAP5 uses mixture momentum equation and difference momentum equation. The neutronic model in this study is the 3D, two group diffusion model (REMARK), which provides realistic thermal boundary conditions.
Figure 4 Separator model

Figure 5 Feedwater flow in scram mode
4. Test results

Three tests are conducted for each nodalization. The first one is a full power steady state initialization. The purpose of this test is to check the momentum balance. Recirculation pumps run back is the second test. In this test, recirculation pumps run down to minimum speed (314 rpm) in 8 seconds. The level is controlled by feedwater level controller. The sensitivity of the level response is investigated. In the third test, a reactor scram from full power is studied. The influence of core power on level performance is evaluated. In this test, the feedwater flow follows the scram mode operation and the steam flow is arranged to be as close to real plant as possible. All three nodalization are subjected to the same steam flow and feed flow conditions.

4.1 Steady state initialization

Nodalization 1 shows good steady state performance. No oscillation is shown in the results. However a substantial deviation of liquid inventory of Node 29 and 2 is shown in Figure 6c. Liquid amount is much less inside separator than outside. This deviation of inventory is difficult to be verified because few data of the mass inventory inside separator is available.

In Figure 7, results show a substantial amount of oscillation in Nodalization 2. The core power oscillates for 3.5% and feedwater flow for 500 kg/s periodically. A closer look at the flow and void plots, the oscillation occurs in the separator. Void fraction in Node 2 and 29 change rapidly and travel in opposite directions. Junction flow 28 is bouncing back and forth the zero line. In a stable case, this flow should be close to zero. This result indicates a force imbalance existing in the nodalization. As mentioned, there are two loops in the nodalization. The first one is the main loop from core to Node 28, Node 29 and Node 3. The second loop is from Node 29 to Node 1, Node 2 and Node 3. It is difficult to maintain a complete force balance between both loops with a common flow - Junction flow 25. This flow depends on the separator model. A small perturbation due to feedwater flow (in Node 3) or gravity difference between Node 2 and 29 could start the instability. This nodalization is very close to real plant geometry, unfortunately is not numerically stable. The solution to this instability might require fine nodes in the separator or a better steam separation mechanism.

The instability caused by the dual loops disappears in Nodalization 3. The result shows perfect steady state response (Figure 8). The drawback of this nodalization is no information of the mass distribution within separator region is available.
4.2 Recirculation pumps runback

Plant data in this transient shows 15 cm swell and 8 cm shrink. Core flow reduces to 4500 kg/s and neutronic power stabilizes around 57 % with lowest value of 33 %.

Figure 9 shows the transient results of Nodalization 1. The level swells 19 cm and shrinks 10 cm. Compared to plant data, the level response is satisfactory. The neutronic power, core flow and pressure are responding in the correct trend with the simplified boundary condition. Core flow reduces to 4200 kg/s. The core power reaches 53 % after transient with a lowest value of 32 %. In Figure 9e, one interesting behavior of the void fraction in Node 29 and Node 2 is shown. As the transient begins, the liquid starts to accumulate in Node 2 due to pumps trip and thus the level increases. The average void fraction in the core increases because of loss of subcooling. The liquid in Node 29 is pushed over to Node 2 by the core voiding. Thus the void fraction in Node 29 increases as average void in the core. It seems the response is over reactive to the loss of cooling. The void fractions at Node 29 and 2 changes at different directions. In real world, as the core flow reduces, the core void will increase due to loss of cooling and the liquid will be pushed up to the separator region. Thus the void in Node 29 should respond in the same direction as Node 2. This behavior has no data for verification.

The feedwater flow responses rapidly shows a large proportional constant is used in the level controller.

The result of Nodalization 2 is interfered by the instability. Thus the average values are used for comparison. Reactor level swells 10 cm and shrinks 4 cm. The core flow drops to 4100 kg/s and feedwater flow drops to 700 kg/s. The core power reaches 50 % after transient and has a lowest value of 30 %. Compared to plant data, the response is correct but not sufficient. Regarding the void fraction in Node 29 and Node 2, they synchronize each other very well. In spite of the instability, they respond in the same direction. In addition, the magnitude of instability decreases as the power reduces to its lower state. This is logical since the instability depends on the feedwater flow and core flow. The results are shown in Figure 10.

Nodalization 3 shows similar behaviors as Nodalization 2 but less responsive. The level swells 10 cm and shrinks 3 cm. The overall response is close to Nodalization 2 except no instability. All the monitored parameters are responding in the right direction. Core flow drops to 4300 kg/s. Core power reaches 53 % with a lowest value of 35 %. This response is expected because the integrated node is much larger than the individual node in Nodalization 2.
4.3 Reactor scram from full power

Plant data shows the lowest level of reactor is 3.25 m for this test and the time to reach lowest level is 40 seconds. The low level (less than 3.8 m) duration is about 50 seconds. Core flow reduces to 3750 kg/s after scram. The chosen plant data is a scram with turbine trip and no dump valves open. The reactor pressure remains higher than the normal manual scram. Thus the reactor level drops lower than the normal scram.

The result shows Nodalization 1 has a large deviation than the other two nodalizations and plant data. First of all, the reactor level drops too low (3.0 m) in a short period and stays low too long (113 seconds for level below 3.8 m). It takes 14 seconds to drop to the lowest level. The core flow drops as low as 700 kg/s and it recovers slowly to 2900 kg/s for 175 seconds. This is due to high flow resistance in the loop. The void fraction of Node 2 and Node 29 show opposite directional response as in the case of pumps runback test. In this case, as the pump head decreases and loss of thermal power, there is not enough driving force to push liquid from Node 29 over to Node 2. This is again an over reactive response to the core void shrinkage. The worst scenario occurs when Node 2 is totally voided and Node 29 is completely filled. In the later stage of the transient, the decay heat warms up the core fluid and the level swells. The bubble generates in the core and travels to Node 29. As the void fraction in Node 29 is greater than zero, the level increases rapidly. The rapid level increase is not physically meaningful for training.

Nodalization 2 shows fairly well response in the test. Node 2 and 29 respond synchronously to loss of power and pump head, shown in Figure 13. The core flow reduces to 3500 kg/s and level drops to 3.5 m. The results are satisfactory compared to the plant data. The duration of low water level (below 3.8 m) is about 26 seconds and it takes about 25 seconds to reach the lowest level. The level (average value) responds smoothly in spite of the instability. The instability reduces to minimum as the transient ends since there is minimum core flow and no feedwater flow is present. Figure 13f shows Junction flow 28 reduces significantly as the transient finishes.

Nodalization 3 shows similar behaviors as Nodalization 2, except no instability. Duration of low level is 22 seconds and it takes 22 seconds to reach the lowest level of 3.6 m. Core flow drops to 3500 kg/s and stays stable at the value. All the parameters respond in the right direction and the trends are the same as Nodalization 2.
Figure 6 Steady state initialization for Nodalization 1 (169.9 seconds runtime)
Figure 7 Steady state initialization for Nodalization 2 (170.8 seconds runtime)
Figure 8 Steady state initialization for Nodalization 3 (174.0 seconds runtime)
Figure 9 Recirculation pumps runback for Nodalization 1 (116.1 seconds runtime)
Figure 10 Rectirculation pumps runback for Nodalization 2 (113.9 seconds runtime)
Figure 11 Recirculation pumps runback for Nodalization 3 (114.6 seconds runtime)
Figure 12 Reactor scram for Nodalization I (238.1 seconds runtime)
Figure 13 Reactor scram for Nodalization 2 (174.4 seconds runtime)
Figure 14 Reactor scram for Nodalization 3 (171.8 seconds runtime)
5. Conclusion

In this study, three tests are conducted to compare the thermohydraulic performance of the test nodalizations. Both quantitative and qualitative results are thoroughly examined. The quantitative results may deviate from plant data because of the simplified boundary conditions. However the boundary conditions do provide a practical environment for comparison. Meaningful conclusions are drawn from these experiments.

For the level response, Nodalization 1 shows responsiveness in the pumps runback test. While in reactor scram test, it is too responsive. The void fractions in the separator nodes do not equalize well during the transient and the result is not realistic. It could cause difficulties for reactor level controller in a simulator environment. Nodalization 2 and Nodalization 3 have overall satisfactory results in the transient tests. However they are less responsive compared to plant data. In terms of stability, Nodalization 1 and Nodalization 3 have stable performance. Nodalization 2 has a generic instability embedded in the steam separator region. The model produces an unbalanced momentum in the conjugated loops which generates this instability. A substantial amount of oscillation is shown in the full power steady state condition. It influences all transient responses as well. Test results also show that the oscillation reduces as the power decreases. It might require fine nodes for the separator region and a more sophisticated separator model to remove this instability.

Nodalization 3 demonstrates a stable thermohydraulics throughout steady state and transient tests. Compared to other nodalizations, it is slightly less responsive. A more realistic boundary condition, such as reactor pressure controller and a sophisticated reactor level controller, could improve the level response. In addition, reducing the integrated node size may further improve the results. Although this nodalization loses the dynamics of inside and outside steam separator, yet the reward of stability and realistic level response are valuable in a real time simulator environment.

References


SIMULATE-3 Core Model for Nuclear Reactor Training Simulators

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ABSTRACT

This paper describes the adaptation of the Studsvik nuclear reactor analysis code, SIMULATE-3, to nuclear reactor training simulation. This adaption to real-time applications permits training simulation to be performed using the same “engineering-grade” core model used for core design, loading optimization, safety analysis, and plant technical support. Use of SIMULATE-3R in training simulation permits simple initialization of simulator core-models (without need for tuning) and facilitates application of cycle-specific core models. SIMULATE-3R permits training simulation of reactor cores with the accuracy normally associated with engineering analysis and enhances the simulator’s “plant analyzer” functions.

1. Introduction

One of the difficult issues facing training simulators is keeping synchronization with changes (usually hardware) which occur at the plant. Hopefully, plant hardware modifications will not occur in each cycle or will be of minor importance with respect to simulator fidelity. However, the need to improve plant capacity factors and economics requires that core design engineers make many changes to optimize fuel assembly design and core loading strategies. Consequently, it is a near certainty that the plant core loading will change (and often change significantly) in each ensuing cycle of operation.

Many training simulators use core models which were developed at the time of simulator installation, and many core models have not been updated as plant core
loadings have changed. This has not gone without notice of licensing authorities (particularly in the U.S.), and recently the Institute for Nuclear Power Operation has issued a report (INPO 96-02) which specifically addresses shortcomings in operator training which have arisen because plant core behavior differed noticeably from that of the training simulator. INPO 96-02 raised concerns about the need for cycle-specific core models in simulator training of plant operators. Some of the changes in modern fuel and core designs which have prompted interest in cycle-specific simulator models are:

**Changes in Fuel Designs:**
- Axial blankets being added to fuel
- Axial enrichment zoning of fuel
- Higher fuel enrichments
- Increased fuel burnup
- New and increased-concentration burnable absorbers
- Larger water holes
- New spacer designs
- Part-length fuel rods (BWRs)
- Smaller fuel pin diameters (BWRs)

**Changes in Core Loading Strategies:**
- Longer cycle lengths
- Larger fractions of fresh fuel
- Lower leakage cores
- Higher soluble boron concentrations (PWRs)
- Long-life control rods (BWRs)

**Changes in Key Core Characteristics:**
- Positive BOL temperature coefficients make PWR start-ups more sensitive.
- More negative EOL temperature coefficients make steam line breaks more severe.
- Small margins-to-limits on heat generation rate increase fuel failure probabilities.
• Larger rod worths increase severity of dropped or ejected rod reactivity insertions.

• PWR xenon transients may be unstable at EOL.

• More unstable channel thermal-hydraulics make BWR decay ratios larger.

2. Advantages of Engineering-Grade Core Models

Most training simulators use simplified (often 1-D) core neutronics/thermal-hydraulics models, which are "tuned" to produce the desired core behavior over a wide class of transients. While it is possible to tune the simulator core models each cycle, the simple core models will always suffer the shortcomings of:

• Inaccurate predictions of key core behavior

• Lack of detailed information within the core

• Large engineering resources required to perform tuning and verification

Consequently, some utilities have started using "engineering-grade" core models directly in their training simulators. Use of the detailed core models in training simulation results in several advantages over historical simulator models:

• Consistent predictions of key core behavior with safety analysis

• Flexibility in modeling all core burnup states

• Availability of detailed information within the core

• Best-estimate predictions for complex transients

• Potential for simulator application in "plant analyzer" functions

and perhaps most importantly:

• Direct and simple cycle-specific core modeling

• Elimination of cycle-specific core model tuning

• Savings of engineering resources needed for tuning
3. Studsvik Core Management System

The Studsvik Core Management System (CMS) has been used by more than 30 utilities in 11 different countries for performing steady-state and transient core design, licensing, and safety analysis since 1985. It has been used to analyze more than 500 cycles of operation in both PWRs and BWRs, with fuel from all manufacturers. Studsvik CMS consist of three major modules.

3.1 CASMO-4

CASMO-4I uses a heterogeneous multi-group solution of the neutron transport equation to generate two-group neutronics data for all unique two-dimensional slices of the fuel assemblies in the reactor core. CASMO-4 allows all bundle designs to be modeled in their true heterogeneous geometry, and it can generate all relevant fuel assembly neutronics data as a function of fuel burnup, rod insertion, fuel temperature, etc.

3.2 SIMULATE-3K

SIMULATE-3K2 uses an advanced nodal method to solve the three-dimensional transient neutron diffusion equation and thermal-hydraulics. The key characteristics of this code are.

**Neutronics model**

- Explicit radial representation of each fuel assembly
- Variable axial nodalization (typically 12-25 nodes)
- Full two-group solution of the neutron diffusion equation
- Fourth-order polynomial representation of flux distributions within each node (this is much more accurate than conventional finite-difference methods)
- Fully-implicit time integration
- Assembly discontinuity factors (ADFs) to treat assembly heterogeneities
- Pin power reconstruction for all fuel pins
- Explicit fuel spacer model
Thermal-hydraulics model

- Explicit radial representation of each fuel assembly
- Variable axial nodalization (typically 6-25 nodes)
- Five-equation, fully-implicit, linear-nodal, channel thermal-hydraulics
- Fully-implicit pin conduction solution with 5-region convective heat transfer model

Nuclear Instrumentation models:

- Explicit LPRM, IRM, SRM models for BWRs
- Flux reconstruction for BWR TIPS (neutron and gamma)
- Explicit ex-core detector model for PWRs
- Flux reconstruction for PWR moveable detector signals

3.3 CMS-LINK

The accuracy of any core model is dependent on the modeling of the feedback between neutronic and thermal-hydraulic parameters. In Studsvik CMS, the CMS-LINK code is used to functionalize CASMO-4 data (which is written to a binary neutronics data library) for the following parameters:

- Two-Group Macroscopic Neutron Cross Sections
- Fission Product Microscopic Cross Sections
- Homogenization Parameters
- Kinetics Data (Betas, Lambdas, and Velocities)
- Decay Heat Data
- Spontaneous Fission/Alpha-n Internal Sources
- Pin Power Distributions

Each parameter is functionalized in 1-D, 2-D, or 3-D data tables versus:

- Fuel Burnup (Exposure)
- Coolant Density (Void)
- Fuel Temperature
• Control Rod Type
• Boron Concentration
• Coolant Density “History”
• Control Rod “History”
• Boron “History”

4. Real-Time Modifications of SIMULATE-3K

The off-the-shelf version of SIMULATE-3K requires too much computational resources to run in real-time on existing hardware. In order to achieve real-time performance, without compromising the accuracy of SIMULATE-3K, a two-fold reprogramming effort was initiated in 1995. These efforts focused on two principle goals: 1) reducing the scalar run-times through better programming and efficient utilization of on-chip and bulk caches incorporated into modern workstations, and 2) achieving efficient parallel utilization of multiple CPUs to reduce overall execution times. The resulting version of SIMULATE-3K for real-time applications is referred to as SIMULATE-3R.

4.1 Scalar Accelerations

Initial analysis of SIMULATE-3K revealed that the CPU requirements by module were:

- Neutronic data evaluation: 28%
- Thermal-hydraulic computations: 24%
- Neutron flux iterations: 25%
- Coupling coefficient evaluation: 15%
- Delayed neutron precursor updating: 5%
- Miscellaneous: 3%

Since 97% of the execution time was spent in only five major modules, each of these modules was re-written and overall runtimes were reduced by a factor of approximately 3.5. Much of the improvement came from restructuring of data to minimize cache misses, particularly in the thermal-hydraulic and flux iterations.
4.2 Parallel CPU Implementation

Parallel CPU acceleration of SIMULATE-3R was achieved by using POSIX threads (inherent process creation utilities which are part of UNIX and Windows NT operating systems) to thread in parallel the computation in each of the 5 major modules:

- Neutronic data evaluation: threaded by axial plane
- Neutron flux iterations: threaded by nodes
- Fission source updating: threaded by nodes
- Delayed neutron precursor integration: threaded by nodes
- Thermal-hydraulic computations: threaded by channel

These modules account for approximately 80 percent of the CPU time required for execution. Speedups for each of these modules average approximately 3.3 on 4 CPU machines (DEC 2100 4/233, DEC 4100 5/400, and SUN UltraSPARC-2) and 1.7 on 2-CPU Pentium Pro machines (200 MHz) with Windows-NT.

Because 20% of the coding (CPU-wise) is not parallel, the net speedups for SIMULATE-3R are limited to approximately 2.3 on 4 CPU machines (DEC 2100 4/233, DEC 4100 5/400, and SUN UltraSPARC-2). The best speedup achieved to date is 5.5 on a 14 CPU SUN UltraSPARC-2.

5. Simulator Integration

The SIMULATE-3R core model is one of many plant models required for complete plant simulation. Consequently, the SIMULATE-3R core model is treated as one of the slave processes under control of the simulator Executive system.

5.1 Executive Communication

Synchronization of the SIMULATE-3R core model with the rest of the simulator is controlled by the simulator Executive, which is responsible for:

- Selection of model initiation (core burnup state, power, flow, etc.)
- Database communication
- Inter-module communications
• Advancement/suspension of simulation
• Creation of backtrack and snapshot file
• Display variable selection
• Error handling

In order to make the SIMULATE-3R core model compatible with Executive functions, a C Language event loop is used so that on-going SIMULATE-3R simulation is suspended (after each frame) to process instructions from the Executive.

SIMULATE-3R is integrated into a simulator under the assumption that SIMULATE-3R and the simulator have separate (not shared) computer memories. Consequently, SIMULATE-3R and the other simulator models can be executed on either the same or different computers, without modification. However, a process of communication (usually RPCs or sockets) between the Executive and SIMULATE-3R is required. The Executive passes messages to SIMULATE-3R to control:

• Initialization of model
• Loading of a snapshot or backtrack
• Advancement to next frame
• Creation of backtrack or snapshot file
• Termination of simulation
• Selection of display variables
• Updating of SIMULATE-3R core inlet conditions
• Updating of control rod positions

SIMULATE-3R passes messages to the Executive for:

• Completion of initialization
• Completion of frame advancement
• Completion of backtrack or snapshot (loading or creation)
• Occurrence of errors
• Updating of display variables
• Updating of core outlet conditions
• Updating of in-core instrumentation signals
• Updating of ex-core instrumentation signals

5.2 Reactor Coolant System Integration

One of the most important technical details in the integration of SIMULATE-3R with an existing simulator is the method used to connect SIMULATE-3R to the simulator reactor coolant system (RCS). It is very important to keep the simulator RCS loops intact so that the normal process of pressure balancing around the loops is not broken. The method used to integrate SIMULATE-3R with the simulator RCS is:

• RCS lower plenum phasic flows, enthalpies, and upper plenum pressure conditions are used as boundary conditions for the SIMULATE-3R core model
• SIMULATE-3R core neutronics and thermal-hydraulics are advanced one frame
• SIMULATE-3R heat fluxes are passed to the RCS model
• SIMULATE-3R core exit flows and enthalpies are used to compute fill junction values for the RCS model

There are two principle modifications of the RCS model that are required for SIMULATE-3R integration. The first modification is that the RCS heat structures are altered so that the SIMULATE-3R heat fluxes are used directly to drive the RCS core channel thermal-hydraulics. This permits the detailed SIMULATE-3R core model to drive the heat addition to the primary system.

The second modification is the addition of upper plenum tees in each thermal-hydraulic channel. These tees are used as fill junctions to force the RCS model to agree with the SIMULATE-3R upper plenum flows and enthalpies. The fill junction values are assumed to be unchanged over a single time step, so that the RCS models can be advanced (to obtain core inlet conditions) before the SIMULATE-3R core model is advanced. Under pseudo-steady-state conditions, the fill junction values are very small. During severe transients (particularly those with large spatial variations in core thermal-hydraulic conditions), the fill junction values become larger because the RCS core model (usually 1-D) does not accurately reflect the 3-D nature of the core thermal-hydraulics. The fill junctions assure that the upper plenum conditions match those predicted by SIMULATE-3R.

5.3 Backtrack And Snapshot Generation

The simulator will usually save backtrack and snapshot data within the formal database that is used as part of the simulator. SIMULATE-3R requires about 4 to 8 M-bytes of
data to store the information associated with the detailed 3-D core model at each snapshot/backtrack. Consequently, it is not practical to save this information directly in the database. Rather, it is much more efficient to let the Executive assign a file name (which is stored in the database) for each desired snapshot or backtrack, and let SIMULATE-3R create a local data file (which is stored on the computer which is executing SIMULATE-3R).

The time required to physically write a backtrack file is about 1-2 seconds, and such a delay is unacceptable during real-time simulation. Consequently, SIMULATE-3R performs a memory copy of the required backtrack data at the time the backtrack save command is received. The data is then split in to roughly 150 equal-size blocks so that a portion of the data can be written to disk at each time step, and real-time simulation is preserved. With this approach the maximum frequency of backtrack file creation is about 30 seconds.

Snapshot creation and loading is performed with the simulation frozen, and real-time considerations are not important. Consequently, the snapshots or backtracks are read directly, which takes about 1-2 seconds.

5.4 Integrated Run-Time Performance

The initial delivery of SIMULATE-3R for training simulation has been made in Japan as part of a GSE (formerly EuroSim, AB) Simulator Upgrade Project. The SIMULATE-3R core models was integrated to three reactor simulators: a BWR-2, a BWR-5, and a 4-loop PWR. As integrated into the simulators and executed on a DEC 4100 S/400, the SIMULATE-3R core model runs about 15% faster than real time on the 764 assembly BWR and about 50% faster than real time on the 193 assembly PWR. This includes explicit modeling of all assemblies in both thermal-hydraulics and neutronics.

6. Core Model Initialization

One of the most significant aspects of the use of SIMULATE-3R core models in training simulators is the ease at which the simulator models can be initialized. Any utility which uses SIMULATE-3 for its core design and/or analysis can simply use existing SIMULATE-3 restart files to initialize the simulator model. All core geometry, core loading, and fuel assembly data is immediately available from the existing restart files. No normalization or tuning of the core model is required for simulator applications.
In addition, existing neutronics data libraries are used directly in the SIMULATE-3R core model, and no additional effort is required to obtain fuel assembly neutronic data (e.g., cross sections, delayed neutron data, etc.)

7. Summary

Serial and parallel re-coding of the principle modules of SIMULATE-3 have reduced runtimes by a factor of 8.0 on 4 CPU machines. These changes have made application of the SIMULATE-3R "engineering-grade" models practical for real-time training simulation.

The delivery and installation of SIMULATE-3R represents the first time that an advanced nodal reactor analysis code (normally used for licensing analysis) has been used in a training simulator to provided detailed (assembly-by-assembly) models for PWR and BWR reactor cores. This represents a significant step forward in the realization of high accuracy simulation and in implementation of cycle-specific training simulation models.

Use of SIMULATE-3R permits reactor core simulation with the accuracy normally associated with engineering analysis and enhances the simulator's application for "plant analyzer" functions.

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ATLAS: Applications Experiences and Further Developments

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Abstract

An overview of the plant analyzer ATLAS is given, describing its configuration, the process models and the supplementary modules which enhance the functionality of ATLAS for a range of applications in reactor safety analysis. These modules include the Reliability Advisory System, which supports the user by information from probabilistic safety analysis, the Procedure Analysis for development and test of emergency operating procedures, and a diagnostic system for steam-generator tube rupture. The development of plant specific analyzers for various power plants is described, and the user experience related. Finally, the intended further development directions are discussed, centering on a tracking simulator, the migration of the visualization system to Windows NT, and the construction of the Analysis Center as a multimedia environment for the operation of ATLAS.

1. Overview

Since 1992, when the basic configuration of the ATLAS analysis simulator was presented at the CSNI Specialist Meeting on Simulators and Plant Analyzers /BER 92/, ATLAS has grown to a multifunctional tool with applications in the field of nuclear plant safety, the construction of plant specific analyzers, accident analysis, development and assessment of accident management and emergency procedures, human factors studies and education.

Fundamental to the developments of the last years was the open design of the ATLAS architecture. This paper offers a review of those developments, the current ATLAS status, the experience with applications and further enhancements planned in the near future.
1.1 Configuration

ATLAS has been redesigned according to client/server principles (figure 1). The data server takes a central position; it administers the resulting data of the simulation models and keeps them ready for all systems that are linked to the network. These systems include the visualization and interactive communication (MONITOR) and supplementary systems such as decision support, diagnostic and surveillance systems. The server stores the updated data as well as the history of all arriving data. Simulation, server, visualization programs and supplementary programs are run as separate computing processes (clients) which communicate via data interfaces. The processes may be distributed to one or more computers within a network. The system actually runs under UNIX, and hardware of different manufacturers can be used (developments towards a heterogeneous environment including Windows NT will be discussed later). Several visualization programs may be connected to the server. Each of these programs serves one or more color graphics screens (X-terminals or PCs with X-emulation). Up to 10 different graphics windows can be opened on each screen. The interface between the simulation models and the server is thus designed that other simulation models can easily be included. The visual display system is supplemented by a graphics editor which creates the images interactively and prepares them for adaptation to the dynamic data.

The transfer of data between the processes has been realized with RPC (Remote Procedure Calls). This basic mechanism for communication between individual processes is available on all UNIX platforms and PCs and provides high flexibility is given regarding network and computers.

![Figure 1. The configuration of ATLAS](image)

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The server decouples the visualization and other systems from the process data. From the history stored in the data server, time functions (trends) of all process data can be retrieved. The connected programs may request every desired date during the entire stored trend. Playback of stored simulations or during the simulation is supported, as well as the restart possibilities of some simulation codes.

Visualization takes place via graphics constructed by means of a graphics editor. The graphics objects may be attributed with the requisite dynamic possibilities. The ATLAS graphics is based on the Graphics Kernel System (GKS), an international standard. For transformation of process variables into visual dynamic effects, all geometrical and graphic attributes of the graphical objects may be varied according to the data obtained from the server (e.g. scaling, assignment of a color index, conversion to text, moving of image objects). Arithmetic operations on process variables may be performed.

1.2 Simulation Models

The model basis for the simulator is provided by best-estimate codes for the thermal fluid dynamics in the reactor coolant loops, and for the physical processes in the containment. The thermal fluid dynamics is modeled by the ATHLET system code /TES 96/. ATHLET is employed for modeling plants of different design as well as experimental facilities. It also includes an environment for the simulation of the instrumentation and control systems of the plant. For description of the processes within the containment, the GRS–codes RALOC and COCOSYS are employed, for the simulation of severe accidents the third–party integral code MELCOR.

The introduction of MELCOR into the ATLAS environment documents the ease of extending the ATLAS model base via an interface that has been kept as universal as possible.

1.3 Supplementary Modules

Applications of ATLAS to a variety of tasks, which is the essence of a multifunctional tool, are made possible by coupling additional modules to the simulation data. Such modules may e.g. help the analyst by means of decision support, implement operating and emergency procedures, diagnose the state of the plant, or couple the simulation models to plant data in order to obtain tracking capabilities. In the following, the systems developed for Reliability Advice, Procedure Analysis and diagnosis of a Steam-Generator Tube Rupture are described. A sum-up of the planned tracking simulator is given in Chapter 3.3.
1.3.1 The Reliability Advisory System

The aim of the Reliability Advisory System RELADS /KAF 94/ is to supply ATLAS users with information from Probabilistic Safety Assessment, thus complementing the ‘deterministic’ simulation information. RELADS, as an example of a decision support system, provides answers to questions concerning e.g. the most probable failure or the most dangerous failure to occur in the actual simulated plant situation. This is achieved by assessing the change of plant risk-level when certain system and reliability parameters change (e.g. component status, failure rates, test intervals), and evaluating the relative importance of different event sequences (based on associated frequencies) and the impact of failures on them. RELADS has been realized by means of the expert system Rtools.

The risk-level change is calculated on-line and presented on three levels: loss of system functions, occurrence of event sequences and occurrence of core damage. As input, RELADS utilizes fault trees, event trees, reliability data and minimal cut sets gained in a plant specific PSA. Figure 2 shows an event tree updated to reflect the path followed by the simulation, displaying the current point in the event sequence, the reachable paths and the contribution of these paths to the top event. In addition to event tree displays, RELADS contains mimic diagrams showing all components considered in the PSA and the components related to the safety functions. Importance measures are displayed textually and graphically. The user has access to all input data and to sorted lists of the components; clicking components will show their properties and offer its failure modes which the user may then select, whereby a recalculation is initiated.

![Event Tree: Small-break LOCA](image)

*Figure 2. Event Tree: Small-break LOCA*
1.3.2 Procedure Analysis

For analysis of Emergency Operating Procedures, a module based on the on-line expert tool G2 has been created and coupled to the data server /IAK 97/. A knowledge base was generated, stating the general knowledge about single procedure steps, their actions and the sequence in which they should operate. Each procedure step has been derived from a general class 'procedure step', leading to a limited number of building blocks. These objects, together with event handling and logic, may be interactively connected to produce the desired procedure. An interface to the data server has been written by utilizing the G2 Standard Interface.

Figure 3 illustrates the main components of the procedure analysis system. In the upper right side of the screen, a control section allows the user to control the simulation and the data interface, and to view all procedures, initiating events, and data. In interactive build-mode, the procedure objects listed in the lower left corner may be connected to form the procedure diagram (lower right corner). In addition to the procedural flow, the safety goals are always displayed in the left upper corner of the screen. The safety goals have been formulated using fuzzy logic, such that an approach to a safety goal may be detected and distinguished from its violation. Additionally, events are recorded (middle screen), and the logic for describing initiating conditions is provided (lower middle screen) to start off the appropriate procedure.

Figure 3. Overview display of EOP's on the expert system screen

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The secondary feed and bleed procedure of a German PWR has been implemented. It may be run in a manual mode, by which the user has to acknowledge every procedure step, or in an automatic mode. In both modes, required actions such as shutting down the main pumps are performed by appropriate procedure steps without manual interaction of the user. The experience so far shows that the plant analyzer represents a very convenient and efficient way for development and validation of EOP's, especially when automatically testing the procedure on many different scenarios.

1.3.3 Diagnosis of Steam Generator Tube Rupture

The module detects leaks in U-tubes of a steam generator by using first principle mass balance equations and the signals currently available in the PWR-plant. Of special interest are the cases when the diagnosis system of the plant based on detection of radioactivity in secondary loops is not effective (e.g., for low reactor power level as during a start of the plant and in case of small leak area). New measurements come every few seconds, and the signals includes:

- primary coolant core inlet and outlet temperatures of all loops,
- system pressure,
- water level in pressurizer and in every steam generator,
- mass gains and losses in the primary and secondary loops for operating systems (make-up, feed water, steam)

The processes are assumed to be quasi steady-state. Liquid and vapor volumes in the pressurizer are assumed to be saturated. As long as the pressure is high enough to ensure one-phase fluid conditions in the primary circuit, the evaluation of the water density on the basis of pressure and temperature measurements is straightforward. The balance equations account for changes in the different parts of the systems and for mass insertions or exemptions as carried out by the corresponding auxiliary systems of the plant. The primary system is considered as consisting of six parts: pressurizer, core, hot leg, two cold leg parts and steam generator primary side. The break area is evaluated by using a critical flow rate module /BER 95/. In the module presented in this paper a leak diagnosis is given on basis of accurate mass estimation and the dynamics of the mass balance in the primary circuit only. Measurements and unbalances in the secondary loops are used to identify the defect loop.
Figure 4. Diagnosis of SG-Leak on the expert system screen
Figure 4 presents the results for a U-Tube leak of initially 10 kg/s after scram at a power level of 1.5 %. The 4 curves represent the time dynamics of the following mass unbalances:

- curve 1 (red): evaluated by the diagnosis system from measurements in the primary circuit,
- curve 2 (blue): evaluated for the primary circuit by ATLAS simulator in a calculation that ends at time $t = 10 \text{ min } 33 \text{ s}$,
- curve 3 (green): evaluated by the diagnosis system from measurements in the secondary side for the defect loop,
- curve 4 (brown): same as curve 3 but for intact loops.

The close location of curves 1 and 2 shows a good agreement between the simulation in ATLAS and the estimation of the mass unbalance module. The form of curves 3 (trend of continuous increase of the unbalance for the defect loop) and 4 (fluctuations of the unbalance around the zero-line for the intact loops) permit an easy identification of the defect steam generator. The module overestimates the break area by 10 % (4.2 cm$^2$ compared to the real cross section area of a U-tube of 3.8 cm$^2$). Checks done in the power range 1-50 % and leak area 10-100 % of a full break show that a reliable prediction of leakage can be done before two-phase conditions occur in the pressure vessel.

### 2. Application Experience

ATLAS is applied in the field of NPP safety analyses both at GRS and external organizations (e.g. technical supervisory authorities, universities etc.). The analyses, in the range of operational transients and accident scenarios including core melting, are performed with the best estimate codes ATHLET, MELCOR and RALOC mentioned before. ATLAS gives support to users in the field of pre- and post-processing of the codes and also in the control of the simulation by interactive means.

#### 2.1 Pre- and Post-Processing

Within the framework of ATLAS a software tool that graphically visualizes input data is available for ATHLET and RALOC. The tool generates nodalization schemes taking into account the geometry data of the thermohydraulic objects, the geodetic height and the defined thermohydraulic network. The user can easily detect input errors by these graphics, e.g. disconnected objects or unreasonable geometry data. The control system
input part of ATHLET may also be visualized in a similar way. The automatically generated schemes may be modified interactively (e.g. position of objects) to achieve the desired layout and may then be supplemented with simulation data automatically for the further use in ATLAS. The colors of the different objects in the scheme refer to the current value of chosen distribution (e.g. void) according to a given color scale (figure 5). Thus the analyst is able to evaluate the local distribution of process parameters with very small effort and also may easily visualize the time trend at any location just by clicking on the desired object.

![ATHLET PWR nodalization scheme](image)

*Figure 5. ATHLET PWR nodalization scheme*

A topic where the scientifically oriented best-estimate codes of ATLAS differ most noticeably from widely used PC products is the file-oriented input description. The requirement of realistic and high-resolution simulation of plant and control systems however leads to very large, unmanageable input files. For the simulation of instrumentation and control systems, an expert-system based input generator has been developed as a prototype, which offers an interactive environment with all analog and logical building blocks available in the ATHLET simulation language. By connecting the building blocks graphically and defining the parameters of the blocks interactively, the creation of control systems is considerably sped up.

The user of RALOC is also assisted in the preparation of input data by an object oriented tool based on Small Talk. The ASCII input file of RALOC is structured by control words (e.g. zones, junctions) and related keywords (e.g. data for the compartments) and
may contain some thousand lines of data that are hard to handle in a text editor. The tool can read a RALOC input deck and maps the data to an internal appropriate object structure with graphical interface classes (input forms) to visualize and edit the data in a more convenient way. After modification the data may be saved as ASCII again. It is planned to extend the tool beyond input forms allowing for the interactive drawing of the object network and storing the data in an object oriented data base for more sophisticated retrieval.

2.1.1 NPP—Simulation

On the basis of ATLAS several very detailed analysis simulators with a large library of pictures have been developed at GRS. The library includes graphics that assist the analyst in understanding the plant behavior, graphics that are available at the plant control room (Siemens PRISCA system), mimics of control- and auxiliary systems for operator interactions and several predefined trend sets with important process values. An example is given in figure 6.

Figure 6. ATLAS graphic „German PWR reactor vessel“

Analysis simulators for German NPPs: Within the frame of the project “Technical Assistance of Suitable Supervision on Licensing of Nuclear Reactors”, sponsored by the Federal Minister for Environment, Nature Conservation and Nuclear Safety, qualified input decks have been produced for NPP specific analysis simulators. These analysis
simulators enable the analyst to investigate in a short time the system behavior for the essential accident paths detected in safety and risk analyses. Up to now four analysis simulators were finished, for the pressurized water reactors (PWR) of Brokdorf and Neckarwestheim (GKN II) and the boiling water reactor (BWR) of Gundremmingen and Krümmel. The simulators for Unterweser (PWR) and Philippsburg I (BWR) are under construction. The simulators are qualified by comparison to measured data of operational transients in the different plants.

**Analysis simulator for a Russian VVER-1000/320 reactor:** Within the frame of the "Program for scientific and technical cooperation with Russia", between the Federal Minister for Education, Science, Research and Technology, and MINATOM (Russia), an analysis simulator for the reference NPP Balakowo, a VVER-1000/230 reactor is being developed. This work is performed by GRS and the Russian partners VNIIAES and OKB Gidropress. The simulator includes the codes ATHLET and CMS (compressible mixture solver) and a library for the simulation of instrumentation and control systems. This analysis simulator will be finished in the mid of 1998.

### 2.2 MELCOR Accident Simulation

For the investigation of core damage sequences with in NPPs the US integral code MELCOR was coupled to ATLAS by small extensions in the executive and control function packages. The MELCOR plot file is used unmodified for the visualization of results in ATLAS with an additional keyword file that describes the plot data with a syntax very similar used in the original plot package of MELCOR. The simulation may be also controlled online with standard simulator commands (e.g. 'PAUSE') and operator actions can be performed interactively by changing the value of any defined control function to a user specified value.

The main emphasis is laid on the interactive presentation of the results of the analyses inside the pressure vessel and the containment. For that purpose a set of special pictures have been created visualizing the phenomena of severe accidents. An example is shown in figure 7, where a PWR cavity is depicted after RPV failure with the red zone (hatched area) indicating the erosion front and thickness of the melt. An important application is the training of shift personnel of the utilities in understanding the course of severe accidents /SON 97/.
2.3 Benefits in using ATLAS

The analysis methods for nuclear power plants are increasingly complex. The input decks for the analysis simulators ATLAS, but also the stand alone input decks for the analysis program ATHLET reach up to 50000 lines. This means that more than one hundred thousand input values are needed for a simulation model of a NPP. The output of the simulation model lies in the same range, with up to 50000 computed values for one time step. The interaction between different systems of a NPP especially for the control systems is also extremely complex.

The evaluation of the high amount of information during a simulation run is nearly impossible by conventional methods using time plots and print outputs. Therefore it is necessary to have a graphical interface that summarizes important information and leads the analyst to more detailed information with synoptically pictures, system representations, sectional views of components and pipes and prepared sets of time trends. These graphical surfaces make it not only easier for the analyst to evaluate and understand the phenomena occurring during a transient but also to explain all the information and results to other interested experts.

With the graphical interfaces the expert can qualify the whole simulation model without changing any data of the input deck only by changing interactive boundary conditions.
for a large number of transients. The correction of detected deficiencies in the input deck will be performed by the specialists who developed the models. In this way the simulation model can be improved by each evaluation of a transient, if necessary. These improvements are then available for the further use of the simulator. Thereby, the database of the simulator is improved continuously and the reliability of the simulator and the calculation results is increased.

With its graphical interfaces the analysis simulator can not only be operated by the developers of the simulator, but also by experts who have a basic knowledge of the system behavior of NPPs and who know the commands of the simulator software. In this way the analyses are more independent of the developers of the simulators. The analyst can interact with the simulation process by the predefined sensitive areas on the graphical surfaces to reduce the probability of operational mistakes. In addition, all interactions with the process model are recorded and can be easily evaluated. In this way the user influence on the analysis results is reduced.

An other advantage of the analysis simulator is that with a qualified simulator model, analyses can be started whenever necessary. No specific know-how of the input deck is needed. Only a short time is necessary to set all initial and boundary conditions for an analysis. This enables the analyst to perform the analyses and the evaluation of the result in a short time with a high reliability.

3. Recent Enhancements and Further Developments

3.1 Visualization and Windows NT

Most of the recent efforts were directed towards improving the user interface of the graphics editor APG used for generating the ATLAS 2D graphics. As an alternative, graphics produced in the CGM format by commercial tools may now be translated into APG. With respect to 3D visualization, plans are to connect a commercial package such as AVS as a client to ATLAS by means of an appropriate interface operating on the data provided by the data server.

The trend in visualization and interactive communication is strongly directed towards Windows NT. ATLAS may be run in a heterogeneous computing environment, enabling a configuration whereby the data server and compute-intensive simulation codes may reside on UNIX machines (possibly with several CPU’s or clustered machines), and the visualization clients on PC’s with Windows NT. First attempts to transfer the ATLAS
visualization to a PC indicate that a conversion to the de-facto graphics standard OpenGL should be readily feasible, the 2D functionality of the OpenGL graphics being comparable to GKS. The strict division of graphics and interactive functions handled by the Windows environment, which is in contrast to the GKS philosophy, will require a partial redesign of the visualization system, but will offer much more functionality as to date.

### 3.2 ATLAS Video Player

As running ATLAS requires well equipped and costly hardware, the ATLAS video player (ATLASVP) was developed, a tool available for all WINDOWS platforms, that allows the visualization of prerecorded simulation runs.

When a simulation of a transient is finished satisfactory, user specified pictures may be recorded by a frame grabbing software to an AVI- or MPEG stream of frames from the running simulator. ATLASVP is able to visualize the recorded streams and can be controlled very similarly to the real ATLAS in off-line mode, but the animation speed is much higher (up to 10 frames/sec). The pictures being recorded have to be chosen carefully, because the user of ATLASVP has no access to the complete simulation data, only to the information recorded in the video streams. Main fields of application of ATLASVP may be conservation of important analyses and the training of understanding the reactor behavior as ATLASVP is based on a multimedia tool and can easily be extended with additional explanations.

### 3.3 The Tracking Simulator

The scope and fidelity of the plant-specific analyzers described before are close to realistically reproducing the plant during normal operating conditions and transients, with exception of severe accidents. However, complete consistency with the plant cannot be obtained because of uncertainties in models and parameters, especially for longer time periods, when errors will accumulate. Combining calculations of the analyzer with plant measurements offers a good chance to account for those uncertainties. From the point of view of safety analysis, this could considerably speed up the time necessary to set up the simulator correctly according to the state of all relevant subsystems and actuators in the plant. Other profitable applications of tracking simulators may lie in obtaining accurate initial conditions for predictions of the plant behavior, e.g. predicting the evolution of transients or the effect of operator actions, and in the availability of otherwise unavailable (not measurable) information for improved information and control systems, including man-machine interfaces.
At the present time a study is being conducted investigating appropriate methods and assessing their feasibility and limitations. As the scope is confined to the requirements of safety analysis, the study concentrates on off-line methods, assuming recorded data; however, most of the methods will as well apply to on-line tracking. Two distinct phases are discerned: first, the steady-state initial phase, followed by tracking a transient phase. For the initial phase, techniques such as uncertainty and sensitivity analysis /KOL 96/ may be used for identifying the parameters strongly contributing to the overall uncertainty, followed by nonlinear parameter estimation. In the transient phase, a concurrent calculation of parameter sensitivities and deployment of recursive least-squares parameter estimation techniques may offer solutions. Another direction of investigation is centered on structuring the entire plant simulation into subsystems, since the requirements for tracking may vary for such systems, e.g. reactor core, fluid dynamics, instrumentation and control systems.

The formal methods however will have to be imbedded in a heuristic framework. This will be obtained by closely following the proceedings already employed by safety analysts performing their work, and casting this knowledge into a knowledge base. In addition, the availability or state of subsystems may be deduced by knowledge based methods. Further methods are of rather technical nature, e.g. transferring measured states of actuators or control signals into the simulation.

Limitations of identification methods are clearly seen when boundary conditions are not known, e.g. leak positions and break areas. Rather, for estimation of boundary conditions, diagnostic tools such as the steam-generator tube rupture system described above should be developed. Thus, the first goal will consists in extending the methodology used in the tube rupture diagnosis to arbitrary primary leaks. In addition, the aspect of signal validity will have an impact on tracking. Signal validation techniques will provide the means to detect and eliminate faulty signals.

As a first step, the configuration of ATLAS has been enhanced to include a second data source (the measured data), and to synchronize the data with the simulation. This is accomplished by means of an 'arbiter' program located in the data stream of simulator data and measurements. The arbiter causes one data source to wait until the other is up to the same time. The flow of combined data is transmitted to the data server, and therefore available to all clients attached. The development of a tracking simulator will be one of the main tasks in the future.

3.4 Analysis Center

The plant analyzer, besides being available on desktops, has traditionally been operated at GRS in the so-called Test Control Room with several CRT's grouped around one
main console. A redesign of the Test Control Room, which will then be named Analysis Center, is under way as sketched in figure 8. The front side of the Center is taken up by a cabinet with two large projection screens lit from behind by LCD-projectors with high resolution and luminosity. The cabinet will also contain multimedia components such as video recording and audio. In front of the cabinet, two cockpit control blocks will be positioned, with four CRT's each. The graphics on each CRT may be directed to either one of the projectors. In the back of the room there will be space for a large conference table, which may be exchanged for seat rows for a larger auditory. A remote control is available for the speaker. The Analysis Center will be suited for presentation and discussion of simulation results, conferences with experts and education.

![Figure 8. Draft Design of the Analysis Center at GRS](image)

4. Conclusions

The architecture of ATLAS has proved valuable in extending the basic simulation and visualization facilities into a multifunctional tool. The interface to simulation codes has allowed to introduce recent developments such as the containment code system COCOSYS and a third-party code (MELCOR) into ATLAS. Although the main applications have been centered on the construction of plant specific simulators for various plants, ATLAS has found applications in procedure analysis, diagnostics and education.
Future developments will concentrate on the issue of tracking. For the safety analyst, tools for setting up the simulation to correspond to a measured plant state will increase the efficiency of his work; other areas such as diagnosis, control system and man-machine design may well profit from this effort. With respect to visualization, 3D graphics will be introduced, and the computing platforms will be extended to include Windows NT.

For purposes of presentation and discussion of simulator results, conferences with experts and education, an analysis center will soon be established, with two cockpits containing four CRT's each, and two large-screen projections.

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SAPHIR, a Simulator for Engineering and Training on N4-type Nuclear Power Plants

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Abstract

SAPHIR, the new simulator developed by FRAMATOME, has been designed to be a convenient tool for engineering and training for different types of nuclear power plants. Its first application is for the French "N4" four-loop 1500MWe PWR.

The basic features of SAPHIR are:
1. Use of advanced codes for modelling the primary and secondary systems, including an axial steam generator model.
2. Use of a simulation workshop containing different tools for modelling fluid, electrical, instrument and control networks.
3. A Man-Machine Interface designed for an easy and convivial use which can simulate the different computerized control consoles of the "N4" control room.

This paper outlines features and capabilities of this tool, both for engineering and training purposes.
1. Introduction

The various uses of simulators in nuclear industry continuously present new needs in real time simulation.
FRAMATOME already had a simulator, built by THOMSON, based on a Gould/Encore 32/87 bi-processor computer. It is used to simulate 3 or 4-loop FRAMATOME designed plants.
In 1991, FRAMATOME decided to develop a new simulator.

There were two principal reasons for this decision:
1. The need for more advanced physical models:
In the existing simulator, the steam generator (SG) model is a simplified model (two nodes) which cannot simulate extreme transient conditions such as complete drying out or overfilling. In order to increase the simulator application range, a new SG axial model has been developed.
To keep the same level of accuracy on both the primary and the SG secondary sides, the modelization of the reactor coolant system (RCS) has been upgraded.
As for the Residual Heat Removal System (RRA), the current model is simple and works only under single phase conditions. It is necessary to have a double-phase model to extend the domain of simulation.

2. The evolution in data-processing domain:
During the past decade, there has been a tremendous evolution in this domain, and there is a benefit in taking advantage of this fact to modernize our simulation tool.
The existing simulator uses the MFX operating system, which is a proprietary system. Most software was developed using the "assembler" language, so that it was not convenient to make modifications.
The capacity of a bi-processor is not enough to take into account the advanced physical models, which need more computation capacity and more memory than the previous models.

When the decision was taken, it was decided that the first application would be to simulate the new N4 model developed by FRAMATOME.

2. Characteristics of the N4 model

The N4 1500 MWe model is characterized by following features, among others:
1. a new SG model 73/19E with improved performances:
   * axial economizer
   * larger heat exchange surface, making it possible to increase the steam pressure
   * more compact design: despite greater performance, the new model is smaller than the 1300MWe model 68/19
2. a new computerized control room, which makes it possible to control the unit in all situations, normal and accidental. The computerized control consoles are entirely dedicated to the reactor operators. They can display the different operating images, computerized procedures, alarm pages and control means.

3. a new reactor control mode (mode X) more flexible, simpler, and more efficient

4. a new reactor protection system with : shorter response time, more flexibility, and reduced amount of equipment.

All these characteristics are simulated in the SAPHIR simulator.

3. Hardware of the SAPHIR simulator

Fig 1 gives an overall view of the simulator. The configuration of SAPHIR is given by fig 2
The simulator is composed of :
1. a real time multi-processor computer Harris Night Hawk NH5800 :
   4 x 2 Motorola processors RISC 88100
   Global memory 64 Mb
   Local memory 4 x 32 Mb
   Disc Memory 4 x 1.7 Gb + 1 x 1.1 Gb
   Real Time clock 250 ns
   VME bus
   Operating system CX/UX UNIX Real Time

2. a man-machine interface with :
   1 supervisor console, including a work station, touch-sensitive screen, 24" color visualisation screen
   4 "user" consoles, including each a work station, 1 touch-sensitive screen, and two 24" color visualisation screens

3. an ETHERNET network connecting computer, supervisor console and "user" consoles.

The software languages are : language C, FORTRAN
4. Simulation model software

4.1. The primary circuitry

1. Modelling
The calculation code used to simulate the reactor coolant system (RCS) is an updated version of the code TRACAS used already in the existing SAF simulator.
TRACAS is a node and junction two-phase model. Phase separation can occur in each node during the transient. The two phases remain at the same pressure but are modelled by separate mass and energy balance equations and can therefore have different temperatures. When one of the phases is saturated, it is subdivided into either water and bubbles or steam and droplets.
Junctions are modelled with one momentum balance equation modified by a two-phase coefficient. Flow splits up into liquid and vapour flows by means of a drift correlation (Zuber-Wallis correlation).
Phase separation models include: bubble rise, droplet fall, wall condensation, interface condensation, jet condensation.
The TRACAS code can handle primary breaks up to 10 inch breaks.

2. Numerical resolution
TRACAS is a five-equation code:
- 2 mass balance equations (two phases per volume)
- 2 energy balance equations (two phases)
- 1 momentum equation (on junctions)
These five conservation equations lead to the equation:
\[ \frac{dX}{dt} = F(X) \]
where \( X \) is the vector: \( X = (\text{Mi}, \text{M}_i, \text{U}_i, \text{U}_i, \text{W}_k) \)
with \( \text{Mi} \) = mass of node \( i \)
\( \text{M}_i \) = mass of gas phase in node \( i \)
\( \text{U}_i \) = total internal energy of node \( i \)
\( \text{U}_i \) = internal energy of gas phase in node \( i \)
\( \text{W}_k \) = mass flow rate in junction \( k \)
The system is then solved, using an implicit method, leading to the equation:
\[ A^* X_0 + B(t + dt) = B X(t + dt) \] is then obtained by reversal of the matrix.

3. Geometrical modelisation of RCS circuitry
The previous version of TRACAS used for 4-loop plants is a model using 24 volumes interconnected by 28 internal junctions. To describe the RCS of N4 plant, the new version of TRACAS uses 54 nodes and 58 internal junctions (see Fig 3). This new modelisation includes following features:
* SG primary side in six nodes to take into account possible thermal stratification in the secondary side. This allows a better computation of natural circulation and its behaviour when the SG is isolated,
* core vessel : 5 nodes. This allows flow crossing between hot legs without core mixture missing, a better description of emptying and refilling of the vessel.
* pressurizer : 3 nodes, one for the surge line to take into account stack effect in energy balance, two for the pressurizer itself to allow thermal stratification in heaters zone.

### 4.2. The Residual Heat Removal System (RRA)

1. **Modelling**
The RRA is modelized by a two-phase model

2. **Numerical resolution**
The same method as in TRACAS is used. However, this method is simplified as follows:
   * all nodes are in thermal equilibrium
   * liquid and vapor flows are not separated
   * no bubble rise is taken into account in nodes

Once the matrix equation is solved, mass and energy balances are performed in each node. Then pressure is calculated by Newton's method.

3. **Geometrical modelling**
The RRA circuitry is divided into 30 nodes with 30 internal junctions (Fig 4).
In order to avoid numerical instability in RCS and RRA coupling, interconnection junctions are treated as internal junctions in TRACAS and their resolution is treated in the implicit scheme.

### 4.3. The SG model

1. **Modelling**
See Fig 5
The model, called GVAXIAL, is based upon a technique of variable volumes. It includes:
* one cold downcomer and one hot downcomer. Each part is divided into: a lower part volume, under the feedwater ring level, an upper part volume which extends to the top level of the steam separators.
* the "riser" part is divided into three regions: one hot region, one cold region, and one mixture region. Hot region and cold region are separated by a partition plate. The axial preheater is situated in the cold region, the boiling section in the hot region. Each region is composed of volumes: 8 in hot region, 8 in cold region, 2 in mixture region
* one "steam dome" Volume

Steam dome volume can disappear in case of overfilling. Vapor can be humid, saturated or superheated.
Upper downcomer volume can disappear if water level goes under feed water ring level.
Liquid can be subcooled or saturated.
Lower downcomer volume can disappear in case of complete steam filling. Liquid can be subcooled or saturated.
Additional riser volume can disappear if tube bundle is uncovered. Liquid can be subcooled or saturated.

This SG model can describe any of the following states:
* Mass from normal inventory up to complete water filling and down to complete steam filling
* Pressure from normal operation up to safety-relief valves actuation and down to complete depressurization at atmospheric pressure
* Flow: all recirculation ratios according to the thermal load and mass inventory

The model integrates the EPRI drift flux model for liquid and gas separation
Recirculation ratio is computed by solving one momentum equation along the recirculation loop.
Mass and energy balances are performed within each loop

2. Numerical resolution

On each variable node mass and energy balances are performed:
This set of semi-implicit equations is then solved by iteration until convergence on total geometrical volume. Then energy and mass redistribution is performed in the different riser zones in order to obtain constant mass nodes.
The downcomer flow is calculated using a momentum equation along the recirculation loop.
Flows in the riser volumes are then calculated using heat flux and pressure loss terms.
Drift correlations are then used to split mass flux into gas and liquid flows which will be used in the next step balance equation

5. Workshop tools for the other models

The workshop used to develop SAPHIR is a workshop designed by CORYS.
The simulation workshop contains several tools:
1. HYDRAULIX, for modelling fluid networks
2. ELECTRIX, for modelling electrical networks
3. CONTROLIX, for modelling I&C networks

These tools are CAD-type graphic component-oriented tools. They support libraries containing objects modelling the different devices of a nuclear power plant.
The common features of these modelling tools are:
- flexibility: all these tools are based on use of object libraries which can be updated and appended by the user. In consequence, maintenance can be done by the user.
- user conviviality: their graphic interface, based on standards, make them easy to use without being a specialist in data processing. After a reasonable period of training, users are able to manipulate these tools for modelling.
- quality of modelling: in physical models, the calculations satisfies global balance (mass and energy for HYDRAULIX, active and reactive power for ELECTRIX)
- performance : the numerical method is optimized.

The three tools are connected to a CORTEX database : this object-oriented data manager enables simulation data to be structured and shared between different software modules. It is also coupled with a graphical tool in order to create and animate images of the Man-Machine Interface. With this interconnection, it is possible to create or to update a database from a document of the modelling tools.

A component of a simulated circuit, either hydraulic or electric, is modelled by a "physical" part and a "logic" part. The "physical" part (thermalhydraulic or electric) is modelled by HYDRAULIX or ELECTRIX, while the "logic" part (or I&C) is modelled by CONTROLIX. The interface between "physical" and "logic" parts is ensured by sensors and actuators. The sensors are output from the "physical" model and input to the I&C model, whereas the actuators are input to the "physical" model and output from the I&C model. Malfunctions are simulated within the I&C model.

The interest of using these CAD-type graphic tools is the reduction of the time required for design. Also the tuning and monitoring of a model can be done without requiring in-depth knowledge in data-processing programming. One particular interest in the use of these tools is that implementation of modifications can be realized within a reasonable time, and incorporated testing functionalities of these tools facilitate validation of modifications.

Fig 6 represents a document made by CONTROLIX.
It can be seen that it is an exact replica of a control sheet of the real plant, which facilitates tracing up to the original objects as well as modifications.

An example of a document made by HYDRAULIX is given by fig 7.

The interest of having a database is that the system is a open system, on which it is possible to connect any kind of supervision tools, which can interact in real time with the database. This also procures a clear separation between models and tools in interface. This feature is particularly interesting, particularly during development stages.
6. Man-Machine Interface

The Man-Machine Interface is designed for easy handling by users following the simulation sessions.

6.1. The Supervisor Console

The control of the simulation session is performed via this Supervisor Console which has all necessary means of control:
* System startup/shutdown
* Choice of plant configuration
* Choice of initial condition status
* Go/Freeze
* Normal, accelerate, slow-down speed
* Replay
* etc...

On this screen, the supervisor has a list of possible malfunctions: breaks, either primary or secondary as well as SGTR, and failure of equipment or instruments. It is also possible to enter macro-instructions to perform conditional malfunctions, to initiate event-triggered, as well as time-triggered malfunctions. At any moment of the simulation session, the supervisor can insert a malfunction. It is also possible to terminate a malfunction as appropriate, to simulate, for example, recovery of an equipment. This allows a wide variety of scenarios, apart from classical scenarios. The touch-sensitive screen allows to perform all these functions quite rapidly.

6.2. The "user" consoles

On the user consoles, the touch-sensitive screens allow the user to perform operations which are in his scope of responsibility. For this purpose, he can have access to:
* all simulated circuits, either fluid, electrical or I&C circuits
* all operable components: to open/close a valve, to start/stop a pump, ...

In order to simulate the computerized control consoles of the actual N4 plants, the displays on the different visualisation screens are, apart from details which are not simulated, the exact replica of images of the N4 plant. By this way, the operator locates the objects, components and disposition of circuitry which is familiar to him.

Fig 8 gives an example of the mimics of SAPHIR (pressurizer).

Apart from these displays, a number of additional didactic displays are available, to teach the physical aspect of phenomenon:
* cross-sections of primary part
* cross-section of the SG's
* logic of reactor control systems,...
The user can have access in real time to the most important parameters: pressure, temperature, etc...
The necessary tools are at his disposal to draw:
- $y = f(t)$ curves
- historic curves
- bargraphs
- operating field curves,...

One of the visualisation screen is devoted to alarms, listed in chronological order. This screen also indicates the moment of disappearance of alarm.

7. Conclusion

Below are some figures to give an idea of the importance of the project:

**Code lines:**
- Man-Machine Interface: 200 000 lines
- Primary model: 140 000 lines
- Other models: 70 000 lines
- Tools: 370 000 lines

**Data lines:**
- I & C data: 900 000 lines
- Fluid system data: 150 000 lines

In conclusion, the SAPHIR simulator, with its open system design, modularity design, conviviality, and advanced thermohydraulic codes, can be used for training operators, in normal and accident conditions, as well as for validating modifications before implementation on site. Its design allows:
- the use of standard tools to develop new models
- retaining the spent software if the hardware has to be changed
- communications with different interfaces

The scope of use of this simulator is large. It can be used, apart from training, to:
1. upgrade existing models
2. investigate the creation of new models
3. validate studies of human factors
4. optimize the control room
5. check disposition taken during the emergency states
6. evaluate accidents in best-estimate conditions,...

In short, a convenient tool for training and engineering.
APPENDIX

1. General overview of the FRAMATOME engineering simulator SAPHIR
2. Hardware configuration of SAPHIR
3. RCS modelling
4. RRA modelling
5. Steam Generator modelling
6. Controlix document
7. Hydraulix document
8. A mimic of SAPHIR
Le SAF
RENOVATION DU SAF

2. HARDWARE CONFIGURATION OF SAPHIR

CONFIGURATION AVEC NOUVEL INTERFACE HOMME MACHINE

CONFIGURATION AVEC PUISSANCE DE CALCUL SUPPLEMENTAIRE

couplage des deux calculateurs
Steam outlet

DOME

ADDITIONAL VOLUME

SEPARATOR

DOWNCOMER UPPER PART

DOWNCOMER LOWER PART (COLD)

RISER (MIXTURE)

DOWNCOMER LOWER PART (HOT)

RISER (COLD)

RISER (HOT)

5. STEAM GENERATOR MODELLING

Primary coolant outlet

Primary coolant inlet

Feedwater inlet
The RELAP5-Based NPA of the VVER Type Paks NPP

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* KFKI Atomic Energy Research Institute, Hungary
** BELGATOM, Belgium

ABSTRACT

NPA [1] is a data driven interactive graphical tool for visualisation of different plant conditions. Data generated by the analysis code RELAP5/MOD3.2 [2] are processed and displayed on a computer monitor. The NPA model of Paks NPP Unit 3 was developed with the aim to demonstrate the phenomena occurring in different transient/accident scenarios. This VVER-specific NPA development is a result of a cooperation between BELGATOM and KFKI-AEKI.

1. Introduction

A Nuclear Plant Analyzer model has been developed for the third block of the Hungarian Paks NPP of the type VVER 440/213.

Data generated by the analysis code RELAP5/MOD 3.2 are processed and displayed by means of the NPA graphical libraries on a computer monitor. Both NPA and RELAP5 are developed and maintained by INEL (Idaho National Engineering Laboratory) on behalf of USNRC and made available via CAMP (Code Assessment and Maintenance Program).

For the existing RELAP5 model of the NPP, the corresponding VVER NPA model realisation proceeds in three phases:

- specification phase,
- implementation phase,
- testing phase.

The result of the first two phases can be described by the resulted masks. The testing phase, which is a longer process is still not finished. Discussion with plant staff is necessary to achieve the most useful layout.
2. Mask specifications

From the point of view of system visualisation the main difference between the PWRs and VVERs is the number of loops that have to be represented. Moreover, the loops are different in the sense of the HPIS, LPIS injections and the location of the pressurizer, therefore the parameters of all the six loops have to be shown. The idea of the organisation of the masks was that there should be a mask showing the general status of the plant. This mask is called the synoptic mask (Fig. 1). The detailed behaviour of the specific parts of the plant or the important phenomena are shown on different masks. These are the following:

- pressurizer mask (Fig. 2),
- vessel mask (Fig. 3),
- alarm mask (Fig. 4),
- loop masks (loop 1 and 6 (Fig. 5), loop 2 and 5, loop 3 and 4),
- steam generator masks (Fig. 6),
- natural circulation mask (Fig. 7),
- loop-seal mask (Fig. 8).

For each mask there is a figure in the paper showing a snapshot of the display.

In order to be able to examine operator interventions, a number of interactive commands have been programmed.

2.1 Synoptic mask

The aim of this mask is to show the overall plant status during transients. From this picture the user can have a global overview of the behaviour of the plant. If something happens in the system the focus can be put on that specific mask to have a possibility of a more in-depth analysis of the phenomenon occurred. Therefore all the modelled parts – primary and secondary side with all the six loops, pressurizer, steam lines, turbines – and equipments – MCPs, HPIS and LPIS pumps and valves, HAs, MCLIVs, MHLIVs, MSIVs, MSHIV, FW/AFW/EFW isolation valves, SG safety valves, pressurizer spray/relief/safety valves, steam dump (BRU-A, BRU-K) valves, turbine isolation valves – are displayed on this mask. For the representation mainly symbols are used to avoid overcrowded outlook of the mask. The main parameters of the system are shown by numbers.

2.2 Pressurizer mask

This mask presents the pressurizer vessel, that allows the user to zoom in on specific thermalhydraulic phenomena that can occur in the pressurizer during transients. The colour in the pressurizer image itself indicates the degree of the subcooling for the liquid phase or superheating for the vapour phase. The three bars on the right side of the
pressurizer vessel represent the collapsed and the measured wide and narrow range levels. During transients the different behaviour of the collapsed and the measured levels can be examined. On the left side the condensation/evaporation rate in the pressurizer subvolumes are indicated. Number components show the parameters characterising the pressurizer.

2.3 Vessel mask

The vessel mask displays a detailed picture of the reactor vessel. In the mask the vessel itself shows the void fraction in the subvolumes of the vessel and the fuel cladding temperature. In the RELAP5 model the core is divided into three channels, so there are hot, 2/3 average and 1/3 average channels. To make clear the location of the hot channel this was put into the middle of the core region and to both sides the 1/3 and 2/3 average channels are placed, symmetrically. On the right of the vessel, deform boxes indicate the hot channel cladding temperatures. The levels are shown on the left, in two kind of partitioning, one for showing the collapsed level for the height of the vessel (vessel – upper head), the other shows a more detailed division (lower plenum – core – upper plenum – upper head) of the vessel. In some transients it can be shown, that the water can be pushed up to the upper plenum, while the core contains mostly vapour. In this case the overall vessel level can show that the water level is above the core. Number components show the parameters characterising the reactor vessel.

2.4 Alarm mask

This mask is intended to easily identify the sequence of events for the transient analysed. On the mask the main protection and safety signals and the status of the main safeguard systems are presented. The boxes indicating the statuses are changing their colour by the corresponding event occurring. For the trip type signals also a number component shows the time of the actuation. The notations of the colours should represent the plant:

- for the pumps and valves the green status shows that the equipment is “closed”, if not, it is red. For example if the feedwater pump is tripped it will be green. (Opposite of the traffic light on the streets.)
- for the protection and safety signals the lamp is off when the signal is false. If the lamp is on the signal is true. The colour of the lamp indicates the danger level of the signal: red is serious (e.g.: scram), green is less serious, white colour means small discrepancies.

In the model the false signal is represented by a neutral colour (light blue), the true signal is red. These colours are used consequently on the other masks too.
2.5 Loop masks

There are three loop masks, each contains a representation of two loops (1 and 6; 2 and 5; 3 and 4). On the masks the primary side temperature distribution is shown by colours in the reactor vessel, the hot and cold legs, the steam generator hot and cold collectors, and in the heat exchanger tubes. In the secondary side the void fraction is indicated in the steam generators. The main loop isolation valves, the MCPs, the charging and letdown valves, the hydroaccumulators, the ECCS systems and the pressurizer can also be seen in the masks. Number components show the parameters characteristic for the corresponding loops.

2.6 Steam generator mask

This mask shows the zoomed picture of the steam generator. In the collectors and the heat exchanger tubes the primary temperatures are shown. In the steam generator the void fraction can be seen. Next to the steam generator the measured narrow and wide range levels are represented. Numbers show the SG pressure, the FW/AFW/EFW and steam mass flow rates.

2.7 Natural circulation mask

This mask is intended to visualise transients during shutdown conditions with low primary parameters.

2.8 Loop-seal mask

The loop-seal mask was created to demonstrate the loop seal clearing phenomenon occurring during LOCA type transients. This mask shows the symbolic view of one loop with both the hot and cold leg loop seals. The levels are indicated in the different part of the loop and in the vessel.

2.9 Interactive commands

In order to be able to examine the results of operator interventions, a number of interactive commands have been programmed. Each can be issued from the synoptic mask. During the planning of the interactive commands we had to notice that only 100 interactive variables can be defined in RELAP5/mod3.2. That is suitable for a three or four loop power plant model, but in case of a sixloop NPP we had to think over the realisation. Table 1. gives the list of the already applied interactive commands. This number is going to increase in the near future by implementing the pressurizer safety valve and extended spray system interactivity.

In the model, all the NPA commands precede the automatic signals. The main role was to reserve the 0 value of the interactive variables for automatic operation. The insertion of the interactive commands increased the number of the RELAP control variables up to 2850.
<table>
<thead>
<tr>
<th>SIGNAL/Equipment</th>
<th>NPA No.</th>
<th>ON</th>
<th>OFF</th>
<th>Auto</th>
<th>Disable</th>
<th>REMARK</th>
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<tr>
<td>SCRAM</td>
<td>1</td>
<td>1</td>
<td>1</td>
<td>0</td>
<td>-1</td>
<td></td>
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<td>BRU-K</td>
<td>2-3</td>
<td>2</td>
<td>1</td>
<td>-1</td>
<td>0</td>
<td>-1</td>
</tr>
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<td>FWTV</td>
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<td>6</td>
<td>1</td>
<td>-1</td>
<td>0</td>
<td></td>
</tr>
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<td>-1</td>
<td>0</td>
<td></td>
</tr>
<tr>
<td>FWP Trip</td>
<td>16</td>
<td>1</td>
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<td>0</td>
<td>-2</td>
<td>disable the trip !!</td>
</tr>
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<td>MHLIV</td>
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<td>1</td>
<td>-1</td>
<td>0</td>
<td></td>
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<tr>
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<td>-1</td>
<td>0</td>
<td>-1</td>
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<td>1</td>
<td>1.2</td>
<td>-1,-2</td>
<td>0</td>
<td>-4</td>
</tr>
<tr>
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<td>37-42</td>
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<td>1</td>
<td>-1</td>
<td>0</td>
<td>-1</td>
</tr>
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<td>4</td>
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<td>-1</td>
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<td>PRZ Spray</td>
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<td>-1</td>
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<td>-1</td>
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<td>0</td>
<td>TU1 trip 1 =&gt; off (X=1,2)</td>
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<td>-1</td>
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<td>-1</td>
<td>0</td>
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<td>1</td>
<td>1</td>
<td>-1</td>
<td>0</td>
<td>-1</td>
</tr>
</tbody>
</table>

Table 1. Summary of the interactive commands

3. Testing of the NPA model

The model has already been tested with a SCRAM and SBLOCA transient. These were given to the plant personnel to demonstrate the capability of the NPA. Three more licensing type cases ( a closure of 6 main steam isolation valves, a 46mm cold leg LOCA and a trip of both turbines ) are ready and given to the plant crew.

There is an ongoing project for the NPP, the procedure of the primary feed and bleed in case of a total loss of feedwater transient have to be analysed. As a first step the following cases will be demonstrated:

- the transient without any operator actions,
- the transient with bleed via the safety valve, which means too rapid pressure decrease,
- the transient with bleed via the relief valve, on which the flow is too small,
- the transient with bleed via the relief valve and its bypass valve, the suggested operator action.

In the second step an interactive input deck will be prepared in order to use for operator training.
We use the NPA for visualising the calculated transients to understand easier the phenomena during accidents. Right now it is used for the calculations for investigation of operator actions in case of large primary to secondary leakages.

NPA is also useful in the EOP development.

Further development of the masks is in progress according to the need of the tests which are to be done and the feedbacks are also taken into account.

**Nomenclature:**

- BRU-K : steam dump to the condenser
- FWIV : feedwater isolation valve
- EFWTIV : emergency feedwater isolation valve
- FWP : feedwater pump
- MCLIV : main cold leg isolation valve
- MHLIV : main hot leg isolation valve
- MSIV : main steam isolation valve
- MSHIV : main steam header isolation valve
- EFWP : emergency feedwater pump
- AFWP : auxiliary feedwater pump
- MCP : main circulating pump
- PRZ : pressurizer
- RV : relief valve
- BRU-A : steam dump to the atmosphere
- TU : turbine
- HPIS : high pressure safety injection
- LPIS : low pressure safety injection
- EOP : emergency operating procedures

**References**

Fig. 3  Vessel mask

CLADDING TEMP (C)

VOIDG

PRESSURE  12.32 MPa
DP VESSEL  249.4 kPa
DT VESSEL  30.0 °C

CLADDING TEMP (C)

Time  0.0 s
SCRAM SIGNALS

1000 \( P_{\text{reactor outlet}} < 11.8 \text{ MPa} \) & prz. nr. lvl. < 0.47 m
1000 \( P_{\text{reactor outlet}} < 9.3 \text{ MPa} \)
1000 More than 3 RCPs tripped with a 3 s delay
1000 \( \Delta P_{\text{reactor vessel}} > 0.37 \text{ MPa} \) & 6 RCPs running
1000 More than 1 RCP tripped & \( \Delta P_{\text{reactor vessel}} > 0.27 \text{ MPa} \)
1000 SG WR LVL < -0.4 m in at least 2 active loops
1000 Turbine trip
1000 MSH rupture
1000 Reactor period < 10 s
1000 Neutron power \( N > 110\% N_{\text{set}} \)
1000 \( P_{\text{containment}} > 0.11 \text{ MPa} \) (BOXH)
1000 Manual SCRAM

Fig. 4 Alarm mask

<table>
<thead>
<tr>
<th>Loop</th>
<th>MCP</th>
<th>MLIVc</th>
<th>MLIVh</th>
<th>FW</th>
<th>AFW</th>
<th>EFW</th>
<th>MSIV</th>
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<td>1</td>
<td></td>
<td></td>
<td></td>
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<td></td>
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</tr>
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<td>2</td>
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<tr>
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<td>6</td>
<td></td>
<td></td>
<td></td>
<td></td>
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</tr>
</tbody>
</table>

PRESSURIZER
SPRAY
HEATERS
PORV
SVs

ECCS SIGNALS

1000 ECCS 1 PRZ nr. level < 0.17 m
1000 ECCS 2 PRZ nr. lvl. < 0.47 m & P < 11.8 MPa
1000 ECCS 3 P < 9.3 MPa
1000 ECCS 4 \( P_{\text{containment}} > 0.11 \text{ MPa} \) (BOXH)
1000 ECCS 6 MSH rupture
1000 ECCS 7 SG wr lvl. < -0.4 m in 2 loops
1000 BOUT Loss of power supply

1000 ECCS
Fig. 5  Loop mask

<table>
<thead>
<tr>
<th>COLOR LEGENDS</th>
<th>Primary Temperatures</th>
<th>Secondary Void</th>
<th>Equipment Status</th>
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<td></td>
<td>300.0</td>
<td>0.25</td>
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<td>0.00</td>
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<td>250.0</td>
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</thead>
<tbody>
<tr>
<td>Mass flow (kg/s)</td>
<td>1472.3</td>
<td>1472.4</td>
</tr>
<tr>
<td>T hot leg (C)</td>
<td>296.0</td>
<td>590.2</td>
</tr>
<tr>
<td>T cold leg (C)</td>
<td>296.0</td>
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</tr>
<tr>
<td>SG power (MW)</td>
<td>229.3</td>
<td>229.5</td>
</tr>
<tr>
<td>SG pressure (MPa)</td>
<td>4.45</td>
<td>4.45</td>
</tr>
</tbody>
</table>
NPA Applications’ Development in the Nuclear Safety Authority Framework

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Abstract

Due to the present tasks of the SNSA (Slovenian Nuclear Safety Administration) there was a need to gain a tool for analyzing the transients of the Krško Nuclear Power Plant at the SNSA. Combining the RELAP5 code with graphical interface NPA (Nuclear Plant Analyzer), the SNSA management saw an opportunity to have a powerful instrument for analyses and assessments on a user friendly basis and without high costs.

The Krško NPP Analyzer is a joint project of the SNSA and the operator, the Krško NPP. The RELAP5/Mod2.5 input deck was constructed by the Krško NPP’s experts and their subcontractors. In 1996 the work started with translation of input model from RELAP5/Mod2.5 version to Mod3.2. This was done by Tractebel which combined NPA masks with translated input deck and constructed new dynamic function and interactive commands between graphical MMI (Man Machine Interface) and simulation code. Since Tractebel performed similar activities for the Belgian plants, their experience was used through a transfer of knowledge to the SNSA.

After this phase of the project, a user of the NPP Analyzer can run accidents as SBLOCA, Main Steam Line Break, Feed Water Break, SGTR, and many other transients activating and combining interactive commands, starting from a full power operation.

This project has not been finished yet. Improvements of the input deck should be done. The Critical Safety Function window will be created.

The analyzer will be a helpful tool during the training program for Accident Assessment Group, which will give to the experts basic knowledge of plant operation, its systems, and physical phenomena during a steady state and transients or accidents. Also a new dimension is added to the existing safety evaluations at the SNSA to confirm the requested level of nuclear safety at the Krško NPP.
1. Introduction

Computer trends lead to an integration of Machine Graphical Interfaces with specific codes for the description of thermo-hydraulic behavior of the NPP systems. Different operational systems can be through software interfaces joined mutually to allow information exchange between different hardware platforms. These improvements enable usage of simulation tools among a broader spectrum of people being involved in nuclear technology. Users need less time to get familiar with the analyses’ tool. Already calculated results can be efficiently presented to other decision making personnel. This results in higher quality of authority decision capabilities and decreased economic costs.

One of the main present tasks of the SNSA (Slovenian Nuclear Safety Administration) is to carefully observe the operation conditions of the Krško NPP and to approve all the safety related modifications. The mainly used operational tool for evaluating the thermo-hydraulic safety criteria demands in Slovenia on the regulatory side is RELAP5/Mod3.2 code which is available to the CAMP members. World practice proved adequacy of today’s best estimate codes to solve technical questions, with a condition that usage of the code should be on the base of understanding its capabilities. Due to the limitations of the number of employees at the SNSA it is not expected to have several full time experts for running RELAP5 applications in the near future.

Combining the RELAP5 code with graphical interface NPA (Nuclear Plant Analyzer), the SNSA management saw an opportunity to have a powerful instrument for analyses and assessments on a user friendly basis. When opening the projects for constructing the NPA applications it was necessary to form basic needs and demands for a supplier. Applications with graphical interpretation of simulated results should be useful for personnel with basic knowledge about nuclear power plant systems and phenomena in it. On the other hand, the SNSA is aware that those applications could not replace full scope simulator training program or basic scope simulators. They cannot be a supplement for a deep understanding of modeling the thermo-hydraulic phenomena using RELAP code.

2. Software and Hardware platforms at the SNSA for the NPA applications

NPA applications run at the SNSA at the Local Area Network via the Ethernet and TCP/IP support. RELAP5/Mod3.2 and NPA 1.3.4 codes are installed at the SUN-ULTRA workstation under Solaris 2.5 operating system and they run interactively. Initially we experienced a few problems with new compiler capabilities.
Figure 1 NPA applications at LAN SNSA

To make NPA applications available to other SNSA personnel a X-Vision Server was installed at SG INDIGO workstation (IRIX operation system), which with X-Vision Client software support at each Personal Computer with Windows 95 platform forms an emulation of graphical X-Windows terminal. With this solution every PC can become a UNIX graphical terminal for running the NPA applications. Those terminals are able to run also interactively independent from each other. Because of the 8 Bit graphic mode of NPA software, we still have some inconsistencies between Windows 95 and X-Vision colors. This is shown only between switching the basic platforms.

Users can print their results directly from NPA applications via hardcopy function to the LAN printers. Snapshots are also available from the Solaris or Windows operation systems. Formally the SUN workstation is used by Nuclear Safety Section, which is responsible for evaluation of all nuclear power plant modifications regarding to nuclear safety. Open technical questions and discussed phenomena are studied first with available models of RELAPS. Runs are made and results prepared for further discussions with other Sections. Figure 1 interprets basic software and hardware platforms at the SNSA.

3. Phases of the Krško NPP Analyzer development

The idea of the Krško NPP Analyzer at the SNSA started in 1994. The Jožef Stefan
Institute of Ljubljana researched the possibilities of analyzer development and defined the basic scope of work. After this, a demo version of an analyzer was made in the frame of the first analyzer's constructing phase. A 2" Small Break Loss of Coolant Accident was modeled with RELAP5/MOD3.1. Basic mask of primary, secondary, and safety system was drawn and dynamically linked with RELAP restart file with NPA 1.3 software. With this demo analyzer a user was able to follow the transient from the beginning up to 5000 seconds in REPLAY mode only.

The project continued in the second phase as a joined aim of the SNSA and the Krško NPP, which contributed a plant input deck for RELAP5/MOD2.5 version. Electrotechnical Faculty of Zagreb was the NPP's subcontractor for main part of modeling. The SNSA performed an independent verification of input deck with a support by an external expert. Through a public bid in summer 1996 a new developer of the Krško NPP analyzer was chosen; Tractebel from Brussels.

First the input deck was translated to RELAP5/MOD3.2 version with standard routines. Some small hand made corrections were necessary. A new stabilization of the input deck was done without detailed tuning. Interactive commands were added for systems and instrumentation which were already modeled. On the basis of working Doel NPP analyzer the developer created different masks containing the Krško NPP modeled systems. Masks were linked with translated and stabilized input deck.

Before installing the Krško NPP analyzer at the SNSA workstation new versions of RELAP5/MOD3.2 and NPA 1.3.4 were transferred from USNRC to the SNSA under an agreement. SNSA staff made successful installation of both software at SUN ULTRA workstation. There was no problem found during file transfer between Tractebel's HP-9000/735 UNIX platform and SUN ULTRA. More detailed description of phase II is described in the next chapter.

In 1997 the third phase of the analyzer project is already defined. The SNSA has a goal to gain a tool which will be capable of:

- graphically presenting phenomena in primary circuit, secondary and auxiliary systems on the basis of defined initial conditions;
- forming a closed integrity of the power plant simulation: primary and secondary circuit, safety systems should be joined in a hierarchical structure of masks;
- containing "as built" data of NPP in the input deck for RELAP5/MOD3.2;
- offering a user friendly MMI supported with a mouse and keyboard, which should allow simple selection of the chosen mask, starting, pausing, and stopping the run, adjusting the proposed input parameters, and interactive commands for further development of a scenario;
- online transfer of the simulated results on the screen through the graphical
symbols and graphs;
- presenting the Critical Safety Functions (CSF) on additional masks;
- printing all the masks and graphs on paper in a presentable form.

In this phase the SNSA has an intention to upgrade the existing input deck for further integration of other interactive commands such as failures of instrumentations, adjustment of control rods and power etc. After the modification of the input deck new stabilization and fine tuning will be done. As an acceptance criteria for simulated NPP’s parameters during steady state and transients ANSI/ANS-3.5-1985 standard will be used, where applicable. Upgrading of the existing Krško Analyzer graphical features will be made with addition of dynamic functions on the old masks and building some new ones. Masks from the second phase will be upgraded with new symbols and interactive commands. Special effort will be made with constructing new masks for Critical Safety Functions, where standard Westinghouse approach will be used.

4. Present status of the Krško NPP Analyzer and main characteristics

After the completion of the second phase, the translated input deck contains changes needed for the MOD2.5 version to be readable for version MOD3.2. These largely resulted in changing the numbers of time step cards, addition of new arguments for defining the volume and junction components, and adaptation of heat structures for MOD3.2 correlations. Part of the modifications was done by using a utility conversion program (ripconv) supplied by INEL. Others were changed by hand. New cards for interactive commands were put in a special file. In some particular modifications like defining the pressure-based level for pressurizer and steam generators swollen level indicators, authors of the input deck were asked for help. More than 700 lines of the input deck were affected by this conversion and upgrade.

A graphical interface between a user and simulation tool is built up by grouped masks, where the main mask presents global, and other masks show detailed presentation of calculated results. Beside this, masks can also be divided to subgroups. The first subgroup presents operation of NPP systems and components. The second one is oriented to clarify some particular phenomena and the third subgroup is an additional group related to important signals and displayed graphs (Figure 2). Interactive commands are available only from the main mask. For all the masks all NPA features are available by choosing an option from the NPA menu.

The Synoptic mask (Figure 3) presents global presentation of the NPP operation
Figure 2  Structure of masks

simulation. It contains an overall view of the transient and plant behavior. Primary and secondary systems are presented through the color driven objects, which are based on a liquid temperature or void fraction. Structure of objects forms model nodalization. Additional numbers and indicators show parameters as flow rates, hot and cold leg temperatures, average and delta temperatures, or main components status (RCP, PORV, SV, pressurizer heaters and sprays).

The subgroup of systems and components is formed by three masks. The first one presents pressurizer and thermo-hydraulic phenomena in it with a degree of subcooling/superheating and evaporation/condensation rate. A user can observe the difference between collapsed and pressure-based liquid level. The second mask contains a reactor vessel, which color is driven by liquid void fraction and fuel cladding temperatures. A cladding temperature profile is on line displayed for estimation of core
cooling level. Secondary system is presented on an additional mask with both steam generators and main components of secondary steam lines. Phenomena’s subgroup is formed of three masks. Whole primary system with coolant vapour/liquid state is put on additional RCS voiding and Loop seal masks. They are constructed for illustration of specific behavior such as single-phase/two-phase natural circulation, reflux condensation mode, and loop seal clearing. The third mask shows conditions in the steam generator, which is damaged in case of SGTR. Four strip charts allow observation in a default time period of 400 seconds. In such a way also gradients of parameters, which can hardly be observed with usual dynamic functions, are now well presented.

The last subgroup contains two masks. An alarm mask is intended to easily identify the precursors and the sequences of events for the transients. ON/OFF functions and numbers are used to indicate conditions of protection and safeguard signals. Status of the main equipment is recognized in the same way. It was never the intention to simulate the control panels from the main control room, but to summarize times of signal activation and to present a global power plant status. Another mask has four dynamic strip charts showing the time history of the plant major parameters.
Interactive commands are one of the most interesting features of the Krško NPP Analyzer. A user can activate operations similar to the operator's. To enable execution of interactive commands new RELAP input cards were dedicated, that can be entered in either new or restart jobs. Every command is based on a variable that can be changed during execution of the runs. This can be done interactively as many times as needed. From the synoptic mask a user can with this commands activate commands such as reactor scram, Anticipated Transient Without Scram, stopping and starting reactor coolant pumps, manipulation of ECCS components, control of valves etc. A set of predefined break locations is added to the input deck in a form of additional valves. Based on the Belgian experience and SNSA feedback the Krško NPP Analyzer can simulate breaks in the cold leg up to 10 inches (Figure 4), different steam generator tube ruptures (from 1 to 10 tubes), guillotine feedwater and steam line breaks.

![Diagram of Krško NPP Analyzer](image)

**Figure 4**  Example of LOCA with the Krško NPP Analyzer

### 5. Usage of NPA applications at the SNSA

A main goal of the Krško NPP Analyzer was to gain a software tool for increasing the knowledge level of the SNSA personnel and for educational and training purposes for other
experts not permanently working for the regulatory authority.

During fulfilment of the SNSA tasks people working at licensing and inspection processes have to be familiar with work which is based on deep understanding of the phenomena in the reactor core and other NPP systems. They should also be aware of the operator’s possibilities for controlling processes from the main control room. Because there is only one nuclear facility in Slovenia producing nuclear power it is by world practice experience questionable to employ operators, who have several years of working experience behind control room panels, to be involved for inspection in the same power plant. SNSA inspectors and staff, who are involved in the licencing process, are trained in the USNRC training simulator courses. This presents time and high costs. To optimize these costs trainees should before taking place in front of full scope simulators panels know theoretical bases for prescribed acts for steady state and transient operation. Knowledge should be built on a comparable or higher theoretical background than the operator’s one. It is not the purpose of the Krško NPP Analyzer to replace today’s practice for SNSA experts to spend a training program on the full scope simulators. But it will be an efficient preparation for this step of their educational and training program.

During the solving of technical problems concerning safety questions at the Krško NPP, a graphical visualization of the opened item can be a useful tool for an analysit to present the theoretical background of the problem to the SNSA decision making group. Concerning as built hardware and software technology at the SNSA this can be performed directly from the computer screen. Calculated transients can be presented with a speed much greater than the real time.

Other potential users of the analyzer are members of the Accident Assessment Group. This is activated during a nuclear accident and other internal or international exercises for emergency preparedness in case of a nuclear hazard. The group is formed of the SNSA’s and external experts, which are during their normal work occupied with jobs outside the plant operation. They usually have knowledge of a specific technical field and they come from Slovenian research institutes and universities. Together with the SNSA staff they form an effective group with extensive scope of experiences. It was discovered that during the early phase of an accident, which could last a considerable amount of time, the members of the group could perform the following tasks:

- To acknowledge themselves about plant state based on it’s key parameters, Critical Safety Functions etc. These information are available through the NPP’s Emergency Response Data System, telecommunication lines via telex, fax and voice, or radio system.
- To predict (based on gathered information and further evaluation) potential evolution of the accident and an amount of release, which is important for further
actions to prevent damaging consequences outside the plant area. Evaluation and prediction is performed continuously, especially after the additional information or any change.

That’s why it is essential for external experts to gain basic knowledge about principals of power plant response characteristics on known initial accident conditions. SNSA Emergency procedures do not predict to use an analyzer in case of an accident. Standard USNRC guidance is used like IRTM [2]. They are based on short instructions and simple formulas for release predictions. To avoid basic errors these procedures should be used by people, who have gained and proved their understanding of evaluated processes on the simulation tools, where the Krško NPP Analyzer can efficiently take place. The SNSA has an intention to organize short courses for these experts, where they could with in advance calculated and animated transients become knowledgeable about the plant responses. It is a goal that experts should continuously develop the knowledge and skills.

In the SNSA Nuclear Safety Department, where the analyzer project is run, it is an intention to prepare a set of specific transients, which will cover main physical phenomena and as many as possible scenarios, which are evaluated in the safety analyses from NPP’s safety reports. Because of a compromise between having an accurate input model and fast simulation, the Krško NPP Analyzer will for the training purposes not run in real time. Transients will be prepared in advance and shown in training courses in REPLAY mode of the NPA software. Already proved CD writable technology will be used to avoid hardware problems with disk space. Such a kind of use together with analyzer’s translated masks in ASCII format is not limited only for the SNSA hardware platform. It can be transferred to different UNIX platforms with installed NPA. Proven couple of RELAP and NPA has shown its benefits in a form of plant analyzers. A lot of activities worldwide are now oriented for severe accident management. In Slovenia it is already proven that coupling MELCOR 1.8.3 and NPA is possible and it works properly [3]. The SNSA sees in this combination another possibility for having an analyzer for severe accidents, which will clarify phenomena predicted with MELCOR code and be an user friendly tool for evaluation of severe accident guidelines and success criteria of probabilistic safety analyses.

6. Conclusions

After installation of the Krško NPP Analyzer at the SNSA workstation and being familiar with the visualized results from the best estimate code we can conclude that:

- SNSA as a regulatory authority is today capable of running RELAP5 and NPA codes without external support. Two SNSA employees are part time engaged in running transient simulation and maintenance of the codes.

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- Until now best estimate RELAP results from the graphs and columns of printed numbers have been shown in dynamic application, where users can direct the transients and in a short time form a result in a user defined form.

- Through the animation of wide range of variables from RELAP output after the second phase we have already estimated the deficiencies of the used Krško NPP input deck, which should be considered in the next project phase of stabilization and tuning. With that, we have proved that NPA can be a useful tool for model improvements.

- Training on the simulation tool has become available also to the experts which will not work as operators or inspectors. This raises the level of knowledge needed for achieving the acceptable level of nuclear safety by world approved standards. In case of accident the SNSA will have a well-prepared emergency group to predict possible releases and with that to generally contribute mitigating the nuclear accident's consequences as much as possible.

The SNSA further sees the possibility to open the project of severe accident analyzer on the basis of coupling MELCOR and NPA.

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CATHARE Approach Recommended by EDF/SEPTEN for Training (or other) Simulators

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SUMMARY
This paper describes EDF’s approach to NSSS thermal-hydraulics: this is the crucial module in a real-time simulator (this constraint relaxes requirements in respect of neutronics) because it determines the simulator’s scope of application. The approach has involved several stages:

- Existing full-scalers (1980-85 design), equipped with a five-equation primary model (about 40 nodes), coupled with a three-equation axial model of the SG secondary side (plus a very simple model for refilling/venting and draining), which can simulate only a small, 2-inch LOCA and up to 15 bar primary-system pressure;

- SIPA(CT) and the new full-scalers at Fessenheim and Bugey (1990-95 design). These tools feature CathareSimu, an outgrowth of CATHARE 1 (six primary-system equations, four secondary-side equations, at least 187 nodes - extended to the steam header -, implicit digital processing, possible parallelisation); this model permits simulation of breaks of up to 12 inches and at very low primary-system pressure;

- SCAR (1995-2000 design) will be adapted from the CATHARE 2 design code (six equations everywhere, non condensables, 2D and 3D modules), and will allow simulator processing of all operating conditions (except for a severe accident, in the strict sense of core melt), including scenarios based on broken primary piping, at atmospheric pressure.

Only the fine-modelling capabilities of CATHARE make it possible to add genuine echoographies to the traditional Man Machine Interface.

1. EXISTING FULL-SCALE SIMULATORS

In the beginning [3], EDF’s full-scale simulators were equipped with one-phase codes (e.g. the first simulator at Bugey 2, which dates from 1977) or two-phase codes of varying homogeneity: the DEF1-2 code, derived from a Framatome code called FraRelap, is a five-equation code for modelling the primary system, supplemented for the secondary system by a three-equation SG-MSS model (9 full-scalers in service at Training Centres at Bugey, Paluel and Caen and on the sites of Gravelines 1-6 and Cattenom 1-4).

SEPTEN, the EDF department responsible for the physical validation of codes, identified their scope as small primary-piping breaks less than 4 inches in diameter, but with a minimum primary-system pressure of 15 bar.

For the reactor startup and shutdown phases, these two models have been replaced by a highly simplified model allowing simulation, without any physical dimension, of the normal operations of Filling and Venting, and of Draining; these are the “patches” that we have christened FV-Ds.

Worldwide, none of the full-scale simulators (i.e. with a lifesize control room) was equipped with better models.
2. THE DEVELOPMENT OF CATHARE

In parallel, the CATHARE\(^1\) project was underway. This consisted of a huge experimental programme on the accident thermal-hydraulics of water reactors: the French nuclear industrialists strove together to understand these complex, safety-crucial phenomena, and to create a code able to rival the American codes TRAC and RELAP. This vast programme resulted in the CATHARE code of which there have been several successive versions:

- **CATHARE-1**, which can be simply characterised as a model of the primary system in six equations and of the secondary system in three equations (the neutronics model is simplified, whether in point or 1D form). This code is now rarely used at EDF, except for general functional studies and for coupling with an advanced neutronic code (CITHARE or COCCINELLE [1]).

- **Cathare-Simu** (C-S) designates several applications derived from Cathare-1 and speeded up, as the name suggests, so they can be run on a Simulator\(^2\).

- **CATHARE-2**, which marks a series of major advances from CATHARE-1, notably the introduction of non-condensables, a six-equation model of the secondary side of the SGs, an axial model of the reactor coolant pumps, a 2D down-comer model, a 3D model for the pressure-vessel bottom head, etc. At present, two versions coexist:
  - **CATHARE-2 V1.3**, used for all current studies,
  - **CATHARE-2 V1.4**, the most recent delivery (01/11/94).

The French nuclear industry therefore enjoys the benefit of a single generic code, whose simulation scope has been validated by several hundred “separate-effect” and “system” experiments on all the Western loops available in the USA, Japan, Germany, etc., up to and

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\(^1\) Code for Analysis of Thermal-Hydraulics during an Accident and for Reactor safety Evaluation.

\(^2\) Cathare-Simu is exclusively owned by the CEA and EDF.
3. FROM SIPA TO SIPACT SIMULATORS

After Three Mile Island, the EDF utility wished to have a version of CATHARE-1 that could be installed in a full-scale simulator [2]. The Cathare-Simu code was completed in 1986 and installed in EDF’s SIPA simulator in 1991, because the computational power required by the code exceeded the ‘Training Centres’ capability.\(^3\)

Of particular interest are the UNIX workstations, with which one can operate the simulated unit (Trainee Operating Desk), lead a training session (Instructor Station) and visualise the physical behaviour of the NSSS via real echographies (Echographic System).

At present, the SIPA simulators are equipped with three main configurations:
- a 900 MW unit, CP1 type (Gravelines 1),
- a 1300 MW unit, P4 type (Pauul 1),
- a 1450 MW unit, N4 type (Chooz-B 1).

The 900MW and 1300MW configurations include a module representing the NSSS (up to and including the steam header), i.e. the Cathare-Simu code adjusted to comply with the “SIPA standards”, and the post-processor providing an “echographic” display of its results.

As simulation requirements are increasingly demanding, both in France (specifications for the CP0 and N4 projects) and abroad (those for Krsko in Slovenia and the German centre in Essen), the current goal is to have a representative code, with no “fine-tuning”, for all significantly two-phase accidents including a large break. Such a code, which has to give highly realistic results, can only be a “design code”.

Moreover, a common approach for design studies and training can potentially generate savings by avoiding costly development (encoding and validation) and ruinously expensive maintenance.

Since 1996 a compact (workstation) version of SIPA providing equivalent real-time performances has been developed: SIPACT.

This has been installed on the 19 French PWR plants, at a university (the INSTM at Saclay) and engineering centres: CIG (at Marseille), SEPTEN (basic design department). This is fully self-contained and therefore far less expensive.

In addition, a new product, SCOOP (a Procedure-based Operation Simulator)\(^4\), has been commissioned at SEPTEN. This simulator of the human operative (and of the operating instructions at his disposal), which was coupled with SIPA N4 in October 1995, is helping to validate the accident operation procedures for Chooz-B 1 NPP.

4. NEW FULL-SCALE SIMULATORS with CATHARE-SIMU

From the outset of the CP0-900 MW project (to create a simulator at Fessenheim and refurbish the one at Bugey) - given that CATHARE-2 was not yet available - SEPTEN recommended using Cathare-Simu instead of DEFI. With the SIMU-N4/1450 MW project (a full-scaler for Chooz-B), the feasibility of parallelising this code was shown and thus permitted the choice of a computation time step less than or equal to 100 ms - an essential criterion for considering rapid transients and activating controls (in normal and incident operation).

The Fessenheim simulator, delivered in March 1997, gave full satisfaction in the targeted normal and accident areas: no dawdling (lagging behind real time) has been observed, whatever the size of the simulated breaks (up to 12 inches in diameter) [4].

These three identical simulators will also include a high-level 1D neutron unit (adapted by the manufacturer, THOMSON, using the LIBELLE/L code supplied by EDF) which among other things ensures easy reception of the neutron libraries used for studying real unit campaigns (load follow, for example). The next phase will be the renovation of all the other EDF full-scale simulators (9 units and 1 new P4) in the year 2001/2002 using the same procedure.

\(^3\) To be more precise, it exceeded the power of the computers of the period: this is why SIPA, which is run on EDF's Cray X-MP in Paris, was installed at SEPTEN in Lyon (SIPA 1). The IPSN did the same with its SIPA 2, which was run on the CEA's Cray X-MP in Saclay, and installed at Fontenay-aux-Roses.

\(^4\) in French: « Simulateur de Comportement d’un Opérateur Observant une Procédure ». Simulator of behaviour of an Operator Observing Procedures.
5. THE SCAR PROJECT

The AFP programme endorsed by the General Management (which addresses Process Training Improvement), seeks to sensitize plant operators to the full spectrum of normal, incident and accident conditions during full-load operation or in open-primary system shutdown conditions.

In France, the only code that can meet this specification is CATHARE-2, whose features (implicit processing, six equations, non condensables) have already been combined. In addition, operation with the primary system open no longer justifies a permutation with such or such a "patch" (for example FV-D).

The SCAR project (Simulator CAthare Release), which is costing about ten millions dollars over a four-year period, consists of:

- producing some physical complements, with the particular aim of ensuring a more reliable response to quasi-atmospheric pressure,
- making CATHARE-2 VI.4 comply with the SIFA standards (refurbished and rechristened CISO) for interfacing with other simulation modules,
- speeding up, and possibly parallelising the "mite" thus obtained, so it can be run on a local computer (of single- or multi-processor type) of the new generations (with shared-distributed memory).

This will place at the disposal of Design Studies, Engineering and Training users, on simulators, an accurate, real-time version of the CATHARE-2 code for all normal, incident and accident operating states, starting from any initial state (See figure 1).

![Simulators Domain](image)

Figure 1: Simulators Domain

6. MORE DETAILS...

6.1 CATHARE-SIMU FEATURES

The first version of Cathare-Simu has the following features (the main simulator adaptations are shown by the symbol △):

- 1 dimension, 2 fluids (6 equations)
- optimal nodalization
- Homogeneous SG model (3 secondary equations)
- Fully implicit quantizing (primary and secondary separated)
• Physical laws and correlations in a separate module (grid), not user-modifiable
• Digital smoothing of the physical laws
• Direct computation of the permanent state (automatic iterative process)
• Finite differences method (staggered meshes and donor cells technique)
• Iterative resolution (Newton-Raphson) and automatic time step management. This choice, at once the weakness and the strength of CATHARE, reflects the intention to prioritize accuracy in relation to calculating speed
• Compliance with real time on the CRAY XMP supercomputer for calculating normal and accident transients with a time step of 500 ms overall
• Validation by comparison with CATHARE-1 on 10 transients plus 6 transients recorded on the reactors and a number of strength tests
• Indication of exit from the physical validity range.

The current versions of Cathare-Simu (for full scope CP0 / N4 simulators and the ones that are planned as part of the renovation of the 900 MW and 1300 MW simulators) include the following improvements:
• Model:
  - Differentiation of steam/liquid velocities and counter-current facility at the SG secondary (introduction of drift/flux correlation)
  - Consideration of thermal inertia of the steam line walls
  - Integration of the economizer (specific to the N4 steam generators)
• Computational parallelization on 4 processors
• Development of a model of transport of up to 16 radiochemical bodies in the thermal hydraulics mesh
• For rapid and complex abnormal transients, e.g. house load, cycle time reduced to 100 ms, whilst ensuring real time on the local computer (SGI R8000 75 Mhz or R10000 90 Mhz)
• Validation by comparison with CATHARE-2 v. 1.3 on 10 transients and with several Chooz-B1 commissioning transient tests.

It will be useful here to give a few figures on the N4: 100,000 lines of FORTRAN 77, 237 mesh links, 80,000 variables. EDF has provided the batch version code; the manufacturer, THOMSON, fitted the interfaces in accordance with the rules laid down for the SIPA Project in order to ensure the modularity of the different simulator software components [2].

6.2 SCAR FEATURES

Keeping on with two CATHARE product lines is not the result of a deliberate policy, but was necessitated by circumstance, mainly by technical limitations in respect of computing capacity (see figure 2).

Today we are heading towards convergence, as (a) probabilistic risk assessments have demonstrated the need to extend operator training and thus the range of simulation in shutdown states (mid-loop operation), a range covered by CATHARE-2; and (b) the rapid increase in computer capacity and the use of parallel computers will ensure a real-time operation despite the increased size of the CATHARE code.

The SCAR Project has just been started up in order to combine, in a single product, applications intended for the analysis and simulation of CATHARE-2. The models developed as a result of this project will then be installed in simulators whose renovation will start soon; and arrangements will be made to set aside the space they will require.

The main features of the SCAR product will be as follows:
• Base - CATHARE 2, v. 1.4:
  - 4 non-condensable gases
  - New modules (1D pump, 2D downcomer, etc.)
  - Six equations for the SG secondary
• Extended scope on the RHR circuit
• Valve or flap valve modules, etc.
• Low pressure digital hardening
• Modelling of the transport of chemical and radioactive bodies
• User friendly pre-processor
• Configurable training post-processor
• Real-time on local computer through code acceleration and parallelization.

Looking ahead to the years 2003/4, these SCAR simulators will cover all normal working operations (filling-venting, draining, etc.), shutdown situations (open primary but closed
RPV) as well as incidents and accidents arising from any of these situations.

Figure 2: From CATHARE to SIMULATORS

CONCLUSION

Equipping EDF full-scale training simulators with an advanced thermal hydraulics code is now standard after more than ten years' work on a specific CATHARE version. Shortly after the year 2000, advances in scientific computer performance will make it possible to restore the single nature of the CATHARE code, both for control purposes and for training [5].

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Use of Simulators for Validation of Advanced Plant Monitoring Systems

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KEYWORDS
Situation Awareness, scope, correctness, simulation, process control, nuclear power plant, performance, satisfaction, alarm masking

ABSTRACT
This paper describes how the full-scope nuclear power plant simulator of Doel (Belgium) was used to assess Situation Awareness for the validation of a process monitoring and supervision system, named DIMOS. The method (derived from a method originally developed for the aerospace industry) has been adapted and applied to compare the efficiency of two versions of the monitoring system: Alarm-masking and non-alarm-masking versions of DIMOS have been analysed in their ability to support Situation Awareness, to improve performance and to fulfil the satisfaction of operators. Both normal power plant operating conditions and abnormal operating conditions were simulated and a large number of power plant operators were involved in the evaluation. The paper focuses on the rationale behind the “Situation Awareness” evaluation, the experiment environment and the results regarding the added value of the alarm masking version of the monitoring system.
1. Introduction

The enormous increase of computer data-processing speeds during the last few years has allowed to develop faster and more complex nuclear power plant monitoring systems. These operator support systems have become part of the standard equipment of power plants and have even become essential for operation. Indeed, several thousands of signals can be scanned several times per second. However, this almost unlimited data-handling capacity leads to an extremely high concentration of information, made available on one or more computer screens for a few operators, possibly leading to 'over-informing' them and also decreasing their efficiency. The supervision system then fails in being an aid to its user. Therefore, it is extremely important to adapt the information to each user's needs in order to give them useful tool which helps them in all circumstances.

**Alarm treatment**, aimed at ameliorating the quality of the process-information, has become the keyword.

Alarm treatment consists in assigning priorities to events allowing important process changes to catch the operator's eye faster than less important ones. Alarm-treatment also means that certain events will not be presented in certain circumstances, and that at other moments they will give an alarm-message on one or more specific operator stations. This mechanism is called 'alarm-reduction', and is aimed at reducing useless information, mainly during avalanches when it is difficult to extract the information which is actually useful.

**Automatic sequence supervision** is another mechanism to reduce the number of messages which might appear during an incident. It synthesizes the results of any automatic sequence (e.g. containment isolation) and verifies the actuator performances or verifies the chronology of the actions and events (e.g. diesel sequence after safety injection signal).

One of the systems that have been developed to enable alarm treatment, is **DIMOS** (Distributed MONitoring System). DIMOS has been developed by BELGATOM and was backfitted at the DOEL nuclear power plant (Belgium). It has been monitoring units 1 and 2 since August 1991, unit 3 since July 1993 and unit 4 since July 1996.\(^1\) It is also used in an other application for monitoring environmental non-radioactive waste. However, together with this new functionality, a new problem arises: how can we test the effectiveness of alarm-treatment? Are we allowed to reduce the number of alarm messages without deteriorating the quality of the information-flow from the process towards the operator? Therefore, it has been decided to validate this functionality on a full-scope training simulator.
This paper describes the experiments that have been carried out on the full-scale simulator at the Scaldis training center in Belgium, starting in June 95, and the results that have been obtained. It also describes the evaluation methodology for comparing two versions of DIMOS (with alarm-treatment and without alarm-treatment) on several normal situations as well as on incidental and accidental scenarios.

2. The Use of a Simulator

It is quite obvious that the validation of a new process monitoring system which pretends to reduce the quantity while increasing the quality of the alarm messaging can not be carried out as an experiment in a real power plant: usually the power plant is in a steady state in which almost no alarms appear. Furthermore one hopes that most of the possible alarms will never appear in a nuclear power plant. Therefore the choice of the full-scope simulator of the Doel4 Nuclear Power Plant was quite evident. The simulator, being used for operator training purposes is a validated tool meaning that the simulation corresponds to real-life situations. Its use opens lots of doors: simulating different scenarios representing more or less severe incidents and accidents in a short time period, makes it the ideal tool for the validation of the new alarm-treatment functionality. Furthermore, the fact that the operators are used to work with it during their regular training sessions also signifies that no special care had to be taken to make them feel comfortable with the simulator.

3. Situation Awareness

The comparison between the two versions of DIMOS is focused on the ability to maintain or enhance Situation Awareness (SA), as well as to sustain performance and provide satisfaction to the operators.

- SA has been chosen as the main criterion for comparing both systems, since an obvious danger with the new alarm-masking version is that it can hide some critical information that normally contributes to SA. SA itself is widely considered as "an essential prerequisite for the safe operation of any complex dynamic system", so that achieving or enhancing SA is a major objective in the design of dynamic process control interfaces.\textsuperscript{2,3,4} The measuring method for quantifying SA is based on an interruptive technique inspired from existing methods.\textsuperscript{3,5} Operators have to provide answers to a set of questions focused on the current state of the process. Neutral (non-pertinent) questions have been added to the set of questions to avoid biases and attentional focusing on the portions of the process under evolution.
As a commonly accepted and integrated definition of SA is still to come, we have simply investigated two different attributes of situation awareness that we considered as important for our purpose: scope measures the range of the SA possessed by operators and correctness measures its quality. To enable measuring of the scope, we had to introduce as one of the possible answers to each of the multiple-choice questions the "no idea" answer. Scope was then defined as the proportion of the ‘No idea’ answers to the total number of questions administered to the operator. Correctness has been evaluated in real-time during the execution of scenarios by a qualified instructor. The complexity of the experiment (variability of the possible process evolution and operators actions) does not make it possible to easily implement an automatic and computerized verification of the answers.

- Performance has been chosen as an other important criterion for comparing both versions of DIMOS because, besides safety, it is obviously a second factor that must be optimized in modern process control situations. Proving that one version of DIMOS is actually better in terms of performance than the other is a critical information for deciding which version must be introduced in a real power plant. Since the use of a simulator makes it possible to also experiment incidental and accidental situations, performance has been measured as the ability to quickly, safely and economically restore the normal state of the process. In particular, it has been rated as an evaluation of the amount of money needed for replacing the hardware or equipment altered, broken or lost during the execution of the scenario.

- Subjective satisfaction of the operators is the third criterion used in this research. Satisfaction usually is viewed as a main contributing factor in usability evaluation methods, and it must be considered as mandatory if one wants to introduce in real power plants a system that will actively be used.

Most measures have been taken in situations where no diagnostic or corrective procedures are supposed to be applied. Maintaining a correct Situation Awareness, both in scope and correctness, is mandatory in these situations for dealing with incidents or accidents. Situations where use can be made of existing procedures have been studied with less involvement since it is considered that SA is less important in these situations where operators merely follow procedures. In such cases however, procedures have been adapted in order to take into account the availability of the DIMOS monitoring system in the control room.
4. Experiment Conditions

As explained before, the main goal of the tests was to find out whether one monitoring system (alarm-masking and non alarm-masking) was 'better' than the other one or not. In order to get the most accurate information, the following decisions have been made:

- tests had to be prepared in close co-operation with nuclear power plant experts and instructors;
- tests had to be done on the full-scope simulator of the nuclear power plant of Doel 4 at the Scaldis training center in Belgium in order to simulate normal operation as well as incidental, and accidental operation;
- both monitoring systems (with and without alarm-treatment) should be indistinguishable, except for the contents and quality of their alarm messages: man-machine interface, mimic diagrams and all other external characteristics to be identical;
- test subjects had to be real power plant operators, who were used to the operation of the power plant, as well as to the DIMOS monitoring system to be tested.

However, several constraints had to be taken into account. There was a poor availability of the operators and of the simulator. Tests moreover had to be done on a small-scale mock-up of the monitoring system: the test results had to contribute to the decision whether investment in full-scale configuration was advisable or not.

5. Experiment Scale

Situation Awareness tests have taken place during 9 -not always subsequent- days. 4 Operator teams or 17 different operators participated to these tests. 29 Scenarios have been 'played', which resulted in 7300 questions, out of which 4640 were used to calculate the SA the other questions being dummies.
6. Experiment Description

6.1 Briefing

Before starting the tests with a new operator team, operators were first briefed about the aim of the tests. They were explained that the test-results were used not to evaluate them personally but to evaluate the DIMOS monitoring system. No information was given on the content of the scenarios, neither on the sequence.

6.2 Questionnaires

At the beginning of the tests each operator team received a dummy questionnaire in order to make them familiar with the type of questions and the method of answering. All questions were multiple-choice questions with one of the options being 'No Idea'. The operator was asked to give this answer rather than to guess if he did not know the real answer.

Questions were asked by means of personal computers in order to facilitate post-analysis. Since the answers always were context-dependent, the instructor had to answer them in parallel in order to get the right answers.

The questions tried to find out the Situation Awareness of each subject in a temporal perspective including the past, the present and the future: some questions checked if the operator was aware of some recent events, others tried to find out whether he had an idea about what could happen in the near future and still other questions just asked direct information on the actual state of the process. All these questions together were meant to build an image of the operator's SA. Special care had to be taken not to influence the operator's SA by means of those questions.

6.3 Test Structure

Each operator team spent 2 days on the simulator for those tests: during one day, the four different scenarios were presented in a random order using either the old (without alarm-treatment) or the new (with alarm-treatment) version of the DIMOS monitoring system. The second day, the same scenarios were presented but in a different order and with the other version of the DIMOS monitoring system than the one used the day
before. Special care was taken in order to avoid that scenarios could be recognized the second time they were 'played'.

At pre-defined moments during each scenario, the simulator was 'frozen' and the test-subjects were asked to leave the simulator control room in order to answer some pre-defined situation-dependent questions. Each interruption took about 6 minutes after which the scenario continued. Those questions had to be answered in a place other than the control room, in order to avoid that information could be taken from the display panels after the question had been asked.

About 3 question-interruptions were planned for each scenario.

At the end of each scenario, the operators' opinion about the scenario, the questions and the monitoring system was asked.

During the tests, all kinds of special events, like unexpected or wrong operator actions were recorded and discussed after each scenario. These records were used to evaluate the operator teams' performance.

6.4 Debriefing

At the end of all tests, all operators were asked to fill out a questionnaire that was meant to get their subjective judgements about the two versions of the monitoring system and about the way tests had been done.

7. Results

Test results were divided into three groups: Situation Awareness, subjective judgement of the operators and performance of the operators.8

7.1 Situation Awareness

As explained before, Situation Awareness was measured using two parameters: scope and correctness. Results show that both parameters always vary in the same way, but sometimes scope improvement is higher than correctness improvement, meaning that the operators think they are much more aware of the process, where in fact they are not.
The variations were analyzed globally over all available data, per operator team, per scenario and finally per operator-role (e.g. primary circuit operator). We took the results for all operators for each version of DIMOS. Assuming that each operator is a representative sample of the population of the users, and assuming that all other test-parameters remained the same for all test-subjects (which obviously is almost impossible), we see that the significance-test on the hypothesis that the alarm-treatment version of DIMOS scores better than the version without alarm-treatment is not significant. This is due to the complexity of the tests which makes that we do not have enough samples to make sure that random events can be neglected.

The following conclusions can be made:

- On a global scale (meaning all data put together) an improvement of both parameters, scope and correctness was detected for the monitoring system using alarm-treatment.

- However, when those data are split-up to find out more detailed information, some remarkable facts can be noted: 3 out of the 4 teams make a very positive score, meaning an improvement with the monitoring system with alarm-treatment for scope as well as for correctness. The fourth one makes a negative score. This last team, however was the only team still using every day (in the Doel 4 unit before installation of DIMOS in '96) an older type of process-computer which mainly differs from DIMOS by the absence of alarm-treatment. After the tests, they explained that since everything was new for them, they even were more lost with the new monitoring system hiding information than they were with the one putting tons of messages on their screens just like their older system.

- During the accident-scenario a notable improvement of both scope and correctness was found, while during the incident scenario the opposite seems to happen. Also, one normal operation scenario seems to give an SA-improvement, while the other does not.

- All operator roles give a better score of SA with the monitoring system with alarm-treatment. One role catches the eye since both scores, scope and correctness are much better than the others': the primary circuit operator (PRIM). This can be explained since this operator is the only one still using the DIMOS monitoring system during incidents or accidents (the others are executing and reading procedures) and he gets much more useful and accurate information at these moments than he did before.
7.2 Subjective Judgement

Two types of subjective information were found: oral information during and after the tests, and written information by means of a formal questionnaire. Both lead to exactly the same conclusions:

Although no information was given on which monitoring system was being used each time, operators could clearly distinguish the different versions, just by watching the quantity of the messages during the scenarios.

All users agreed that the reduced information quantity (by masking irrelevant information) is the only way not to get overwhelmed by hundreds of messages which will never be read and which make the monitoring system useless for incident or accident cases.

The improved quality of the information makes the operator feel much more comfortable in his corrective decision making process, leading to faster and better decisions. Pre-treating information can help operators to remind to do some actions which otherwise and especially in stress-conditions might be forgotten.

7.3 Operator Performance

Finally, a remarkable improvement of performance was detected using the monitoring system with alarm treatment. While several important and expensive (virtual!) equipment broke down when use was made of the system without alarm-treatment, the equipment was saved during the scenarios using the new version, since it gave clear and correct information at the right moment and at the right place.

8. Conclusions

The use of a full-scope simulator was a key-element for the validation of the alarm-treatment in DIMOS. The interaction between the operators and the monitoring system could be assessed not only in normal operating conditions, but also during incident and accident scenarios. Both types of scenarios were used to collect information on how this new functionality would be accepted in a real nuclear power plant and how it could contribute to enhance performance and safety.
One can conclude from the results of the validation on simulator that the alarm-treatment mechanism helps to enhance Situation Awareness and improves performance, by allowing operators to focus attention on the most important information. This is confirmed by the subjective opinion of operators: the alarm-treatment version of DIMOS is preferred to the other version.

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References


CAMS as a Tool for Identifying and Predicting Abnormal Plant States Using Real-Time Simulation

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Abstract

CAMS (Computerised Accident Management Support) is a system that provides assistance to the staff in a nuclear power plant control room, in the technical support centre and in the national safety centre. Support is offered in identification of the current plant state, in assessment of the future development of the accident and in planning mitigation strategies.

CAMS is a modular system, where several modules perform different tasks under the control and supervision of a central knowledge based system, which is responsible of the synchronization and the flow of information through the activated modules.

A CAMS prototype has been tested by the Swedish Nuclear Inspectorate during a safety exercise in Sweden in 1995, with satisfactory results.

Future developments include automatic control of the Predictive Simulator by the State Identification, for the generation of possible mitigation strategies, and the development of an improved user interface which considers the integration of the system in an advanced control room.

CAMS is a system developed as a joint research activity at the Halden Reactor Project in close cooperation with member organizations. The project, started in 1993, has now arrived to the second prototype version, which has been presented and demonstrated at several seminars and workshops around the world.
1. INTRODUCTION

CAMS (Computerized Accident Management Support) is a system that will provide support in normal states as well as in accident states. Support is offered in identification of the plant state, in assessment of the future development of the accident, and in planning of accident mitigation strategies. (It does not give support in execution of the chosen mitigation strategy.)

We imagine different types of users: operators and shift leaders in the control room, the staff in the technical support centre (TSC), and people in the national safety authorities. These different types of users need different types of support.

CAMS picks up information from the plant and transforms it into a more digestible form before presenting it to the users. This transformation process can be controlled by the user.

CAMS consists of a data acquisition module (DA), a signal-validation module (SV), a tracking simulator (TS), a predictive simulator (PS), a state-identification module (SI), a probabilistic safety assessment module (PSA), and a man-machine interface module (MMI). The work of these modules is coordinated by a module called the system manager (SM). In addition, there are the strategy generator (SG) and the critical function monitor (CFM), these two are not integrated into the present version of CAMS.

The purpose of the prototype is to study how advanced information techniques can be utilized efficiently in accident management. Various methods are tested. The possibilities and also the difficulties of the chosen design are evaluated.

CAMS has been tested at a safety exercise at the Swedish Nuclear Inspectorate in May 1995. This work has been reported in a joint paper between staff from the Swedish Nuclear Inspectorate and staff from the HRP, Reference [2]. The main conclusion was that CAMS was useful already in the incomplete form that it had at that time, and that it had room for improvement and extension.

A lessons-learned report has been written, Reference [3], about on-line simulation and estimation, to review ideas that can be used in CAMS. Some of these ideas have been used in the TS.

Emphasis has been placed on making a general design and structure facilitating easy maintenance and adaption to different reactor types. Although specific plant knowledge is implemented inside each module, it should not be necessary to rearrange the whole design when changing to another plant.
2. THE STRUCTURE OF CAMS

Figure 1 shows the main modules of CAMS and the data flow between them.

First we note that CAMS is an information system, data flows from the plant to the user. You can influence what goes on in the modules close to the Man-Machine Interface, but you cannot operate the plant through CAMS.

The plant data are picked up by the Data Acquisition module (DA). From there they flow to the Signal Validation module (SV). Validated data flow to the Tracking Simulator (TS) and to the State Identification module (SI). The TS augments the measurements with three sorts of extra data:

- data which the user should like to know, but which are not measured,
- data that are used for initialization of the predictive simulator, but which are not measured, and
- data that are measured, but which also can be calculated from independent measurements.

In the latter case the TS acts as the calculation assistant of the SV. The cooperation between the SV and the TS is indicated by the double arrow between them.

From the TS, validated and augmented data are available to any module that may request it. In addition to the SV already mentioned, the main customers for such data are the Man-Machine Interface (MMI) and Predictive Simulator (PS).

When requested by the user, the PS will pick up the current state from the TS. The PS may be asked to predict what will happen if no intervention is carried out, or if a certain sequence of interventions is carried out.

The SI produce qualitative information about the plant state: there is or is not a leakage, a component is or is not available, etc. This information is communicated to the user by the MMI. It is also the starting point for the analyses done by the Probabilistic Safety Assessment module (PSA), the Strategy Generator module (SG), and the Critical Function Monitor (CFM).

As indicated, the SG and the CFM have not been integrated into the present version of the prototype.

Data from all these modules flow to the Man-Machine Interface (MMI) to be examined by the user. The user can control the data transformation going on in the PS, PSA, etc., this is indicated by the arrows going backwards from the MMI to these modules.
Figure 1. CAMS main components and the functional links between them

3. THE SYSTEM MANAGER (SM)

3.1 Purpose

The SM (which is not shown in Figure 1) is the common functional interface to all the CAMS modules. The main tasks performed here are:

- Functional linking: All the modules communicate with each other only through the SM.
• **Synchronization:** Activity in different modules must be synchronized to produce meaningful outputs. This is particularly true in this system where the modules operate concurrently.

• **Switching:** Every module can be switched on and off without limiting the operation of other modules.

• **Monitoring:** The SM controls the activity and the flow of information through all the modules. A dedicated SM display has been designed, to be used by the CAMS supervisor on site.

The SM has been developed using the real-time expert system shell G2, by Gensym Corp.. Reference [5].

### 3.2 Description

Figure 2 shows the logical connections among the CAMS modules and the SM. External processes (external to G2) used by each module to perform a task, are represented in circles. In this diagram the modular nature of CAMS is emphasized: each functional module has the same structure and all of them work concurrently under the supervision and coordination of the SM. Basically, each module has the following building blocks:

*Figure 2. System Manager connections*
• an external process, performing the main task (for example a Picasso-3 process for the MMI module)

• a G2 module, that drives the external application and receives the results,

• a communication block (dac, svc,...), which contains the data that must be shared with the SM.

4. THE SIGNAL VALIDATION MODULE (SV)

4.1 Purpose

The Signal Validation module (SV) in CAMS is currently based on neuro-fuzzy techniques. The complete algorithm and procedure can be found in Reference [12].

4.2 Description

Figure 3 shows a simplified diagram of the neuro-fuzzy model. The possibilistic fuzzy classifier is used to identify one or more possible regions of the process operating point (as defined by the set of signals to be validated), to which the incoming sample could belong. The possibilistic nature of this classifier results in a prompt detection of patterns outside the module training volume (which can introduce unacceptable errors in the neural networks response).

The artificial neural networks (ANN) have been trained, each in a different region, in the set of the possible regions identified by the classifier. They work concurrently during the validation process and their output is averaged using the fuzzy membership value of the incoming pattern in each cluster (region). In this implementation, seven clusters have been identified, which cover the entire power-flow map of the reactor.
5. THE TRACKING SIMULATOR (TS)

5.1 Purpose

The purpose of the TS (Tracking Simulator) is to calculate:

- Quantities that are not measured, but which the user should like to know. Used by: MMI.

- Quantities that are not measured, which are to be used as initial values when predicting what will happen. Used by: PS.

- Quantities that are measured, but which can also be calculated from other independent measurements. Used by: SV.
5.2 Requirements

For this prototype, the required estimates are:

- Cladding temperature,
- Relief valve flow,
- Steam leakage inside the containment,
- Water leakage inside the containment.

More requirements may be added to this list later, for instance:

- Water level in the reactor tank (for the SV).
- Steam leakage outside the containment,
- Water leakage outside the containment.

Further, some estimates may be available at no extra cost.

5.3 Description

There are several tools available for building simulators, but they all appear to be made for building predictive estimators, rather than tracking ones. The facilities they offer are essentially this:

They have a set of ready-made components like pipes, valves, pumps, turbines, reactors and so on. You can describe the components of your plant by giving parameters to these ready-mades. I have a pipe, you say, the tool gives you its pipe component and ask you to describe your pipe. Its length is this, you say, and its diameter is that.

You then describe the topology of your plant. The water running out of this tank runs into the pipe, the water running out of the pipe runs into that valve. The modelling tool will establish and solve the corresponding algebraic equations.

Finally you give the initial conditions of your plant and ask what the situation will be a certain time from now. The tool will establish and solve the corresponding differential equations.

No suitable tool for making tracking simulator seems to be available on the market. How can we turn a predictive simulator into a tracking simulator? What we want is an estimation tool. And estimation has a mathematical relationship to prediction.
We receive new measurement data at regular intervals \( t = k\Delta t \). But not all the interesting state variables are measured. We arrange all measured variables \( y_1, y_2, \ldots, y_L \) into a vector \( y \), an arbitrary one of these measured variables is denoted by \( y_l \). Similarly, we arrange all non-measured variables \( z_1, z_2, \ldots, z_M \) into a vector \( z \), an arbitrary one of these non-measured variables is denoted by \( z_m \). Finally, all parameters \( p_1, p_2, \ldots, p_N \) that shall be updated are arranged into a vector \( p \), an arbitrary parameter is denoted \( p_n \). The complete state vector \( x \) is then the collection of all these:

\[
x = \begin{bmatrix} y \\ z \\ p \end{bmatrix}.
\]

(1)

Here, \( 1 \leq l \leq L \), \( 1 \leq m \leq M \), and \( 1 \leq n \leq N \). The number of adjustable parameters \( N \) should preferably be much smaller than the number of measurements \( L \).

Non-measured and measured variables are for instance temperatures, pressures, positions, velocities, quantities for which we can establish equations describing the main part of their development with time, even though our knowledge is never precise.

Parameters are quantities that normally do not change, or at least change only slowly, like lengths and cross sections of pipes, the thicknesses of steel walls and electric resistances. The existence or non-existence of a leakage also is included in this category. These latter things, which we often call "constants", may also change, but we are not able to predict how, or we decide not to do so even if we could. We shall assume that a parameter does not jump wildly around, it displays some sort of continuity. Rather than making the parameter directly a Gaussian stochastic variable, we make its change between one measurement and the next a Gaussian stochastic variable. This does not mean that parameters cannot change, only that we cannot predict how it will change. The entire time development of these quantities is described by a noise term.

The classification into variables and parameters is not always obvious. If we for example enlarge our description of flow phenomena to also include the corrosion of a steel pipe, and establish equations for the change of the wall thickness with time, the wall thickness will be a variable rather than a parameter.

Given initial values for estimates of \( y \), \( z \) and \( p \), a prediction tool can tell you the corresponding estimates one time interval later. The \( p \)'s are trivial, of course, as the prediction will be that they have not changed.

Then the measurements at \( t = (k+1)\Delta t \) arrive, and we have new values for the \( y \)'s. They will be similar to the estimated \( y \)'s, but not identical. The differences between measurements and estimates are called residuals or innovations and they are denoted by \( r \). They represent
what we have learnt by the measurements, the amount of surprise. We can use them to improve our estimates of the measured variables $y$, the unmeasured variables $z$, and the parameters $p$.

In each time interval the state vector is therefore changed twice, by the equations of motion which are handled by the prediction tool, and the updates based on the residuals. This is illustrated in Figure 4.

Figure 4. The state vector is modified alternatively by the equation of motion and the update equation. The update equation is fed from the residual equation.

6. THE STATE IDENTIFICATION MODULE (SI)

6.1 Purpose

The purpose of the State Identification (SI) module is to identify the state of the plant and to communicate information to the user and to other CAMS modules. In the current version, the outputs of the SI module are the following:

- output to the user: textual information describing the plant state, status of critical safety functions and availability of safety systems (see the MMI chapter for more details).
• output to the TS: presence of a steam or water leakage inside or outside the containment and the occurrence of steam release through relief valves.

• output to the PSA: the occurrence of initiating events and the availability of front-line and support systems.

6.2 Description

The state identification module is a knowledge based system with the classical main components: a knowledge base and an inference engine. It is developed using a specific tool: GPS (Goal Planning System, Reference [8]) which provides the inference engine and the knowledge-acquisition tool.

The current knowledge base contains 3 parts:

1. Safety objective trees

State Identification is based on Safety objective trees as defined by the NRC, Reference [13].

The safety objective trees identify the relationships between plant safety objectives, challenges to the safety objectives, mechanisms that cause the challenges and strategies that would mitigate or prevent the mechanisms using a hierarchical tree structure. We have chosen to represent safety objective trees, from the safety objective level to the mechanism level. The strategy level will be the purpose of the future SG module.

So far, we have implemented the trees related to the following safety objectives:

• prevent core damage,

• maintain containment integrity.

2. Leakage detection and steam release

The current knowledge basedetects leakage inside and outside the containment by monitoring the evolution of the containment pressure and temperature and the water level in the reactor vessel.

Steam release through relief valves is monitored by checking the position of the relief valves but also the temperature in steam relief lines to prevent errors due to wrong information about the position of the relief valves.

3. Safety systems availability and initiating events

The safety systems monitored by the SI module are automatically started under given plant conditions. When the starting conditions of a safety system are fulfilled, the SI module checks
if the system is really working and if not, it tries to identify the reason why the system does not work.

For example, when the starting conditions for the low pressure cooling system are met, the flows in the low pressure cooling lines are checked. If there is no flow, the system is declared unavailable. Then the speed of the pumps and the position of the related valves are verified and the user is informed of abnormal situations.

The considered initiating events are LOCA of different sizes (large, medium, small), manual or automatic shutdown of the reactor and loss of important functions like heat sink, feedwater or external power. The occurrence of these events is already detected in the previously described parts of the knowledge base. The information is transmitted to the user and to the PSA module.

7. THE PREDICTIVE SIMULATOR (PS)

7.1 Purpose

The purpose of the Predictive Simulator is to tell the CAMS user what the future state of the plant will be, by running a model of the plant faster than real time. In the case of an accident, the user will see if the safety systems of the plant are sufficient as the accident evolves, or if some interference is needed to reach a safe state. If a safe state cannot be reached, the simulator will give indication of when the plant reaches a critical state.

When the user wants to run a prediction, he will initialize the simulator with the current state of the plant (which is a major task of the Tracking Simulator). The user can then let the simulator run by itself to see what will happen if no mitigation strategy is tried, or by manipulating the controls of the simulator he can test different strategies to see which effect they have and choose the better one.

The simulator can also be used to check what might happen before it happens (if ever). Say, we have lost all auxiliary feedwater pumps but one. If the last one should also fail, how much time do we have before the core is uncovered? With this information the user can be prepared if so should happen (this sequence of events happened at the safety exercise at the Swedish Nuclear Inspectorate in May 1995, see Reference [2]).

7.2 Description

No modules use the output from the simulator. The output goes directly to the relevant predictor pictures and the trend system within Picasso-3. But the control of the simulator goes
through the System Manager. In this way the System Manager knows the commands to the simulator, without being loaded with all the display data.

Operating the simulator is done using three pictures, “Predictor Control”, “Predictor Panel” and “Predictor Setup”.

“Predictor Control” is similar to the process picture, which gives an overview of the process, regarding layout and placement of components. This has been done so that the user will recognize and be familiar with the picture during an accident. There is one important difference; in “Predictor Control” one can control the Predictive simulator. This is done using a mimic style interface where one can point and click on components. A small window will pop up and the state of that component can be modified (a pump can be started, a valve opened etc.).

“Predictor Panel” has a more traditional layout with buttons and sliders to mimic what operators are used to from the control room. This layout gives a more detailed control of the systems in the plant, like manual scram and suppression pool cooling.

“Predictor Setup” is used to start, stop and initialize the predictor. Start and stop are self-explanatory, and initialize is used to make the predictor reflect the state of the plant. Normally initializing will be activated when the plant has changed its state, or when the user wants to try a new mitigation strategy. It is not necessary to stop the predictor to control and modify components, it can be done while running.

8. CONCLUSION

This report shows the actual status of the CAMS project. The first phase of the project (1992 - 94), which is described in Reference[1], [2], and [3], focused mainly on the following tasks:

- information needs during normal and accident conditions in a NPP,
- methods that can be successfully applied to a CAMS system,
- MMI and human factors requirements.

In the second phase of the project the development of those ideas into a working prototype has begun. Its main purpose is to test those methods in a simulated environment, to verify that the many developed functions, using different techniques, can work together producing the desired result in an efficient way.
REFERENCES


[6] APROS has been developed by IVO and VTT in Finland.


[11] ORBIX is a trademark of IONA Technologies Ltd.


AN INTELLIGENT DIAGNOSTIC AID (IDA) BASED UPON THE SIMULATED AND OPERATIONAL EXPERIENCE

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ABSTRACT

A diagnostic system has been developed addressed to timely trouble shooting in process plants during all operational modes. The theory of this diagnostic system is related with the usage of learning methods for automatic generation of knowledge bases. This approach enables the conversion of “cause→effect” relations into “effect→possible_causes” ones. As applied to technical diagnosis, this allows one to generate troubleshooting rules of the kind “symptoms→possible_diagnosis” through “failure→symptoms” relations. The problem of diagnostic rule generation is, thus, reduced to obtaining samples of the symptoms, that is the well known problem of failure modelling. The diagnostic rules are derived from the operation of a simulator according to the following procedure:

- identification of all initiating events and of the corresponding operational modes;
- simulation of all the selected transients for a time sufficient to identify the uniqueness of the plant response under each of the initiating event;
- automatic generation of the set of diagnostic rules that, through the use of temporal logic and a filtering Expert System, associate the evolution of the process parameters and their derivatives to the initial perturbation.

The plant model running under the simulator is built-up by means of the LEGO code. The LEGO code is a modular package developed at the Research and Development Department of the Italian National Electricity Board (CRA-ENEL) to facilitate modelling of the dynamics of fossil-fuelled and nuclear power plants. The LEGO code consists of a library of pre-programmed, pre-tested and pre-validated modules, that represent power plant components and a master program which allows the user to build-up a model by automatically interconnecting the modules in the arrangement determined by the modeler. The reference plant is Sampierdarena power station, that is a combined cycle plant dedicated to produce both electrical and heat power.
The following types of failures can be detected: leakages, cut off or deterioration of the characteristics of flow/pressure sources, pipeline fluid conductance disturbances, excessive passage of the medium through pipelines, defects of control systems of auxiliary mechanisms, sensor defects, damage of equipment elements. The identification of malfunctions is performed through the use of the fuzzy set logic algorithms. The hardware configuration of the prototype system is made up of a network of a Hewlett-Packard workstation and a Digital VAX-Station. The diagnosable failure were 41 items of the following types: partial reduction of pipes hydraulic resistance, increased pipes hydraulic resistance, "on-off" behaviour of flowrate variables, pump switch off, tank leaks.

The paper reports the theoretical background on which IDA is based, as well as a rationale of such approach for diagnostic and some evidence of its effectiveness through an application to Sampierdarena 40 MW cogeneration plant. Finally an outline of an ongoing application to a VVER-1000 plant simulator will be given.

1. LEGO Code Overview

The LEGO code is a modular package developed at the Research and Development Department of the Italian National Electricity Board (CRA-ENEL) to facilitate modelling of the dynamics of fossil-fuelled and nuclear power plants. The LEGO code consists of a library of pre-programmed, pre-tested and pre-validated modules, that represent power plant components and a master program which allows the user to build-up a model by automatically interconnecting the modules in the arrangement determined by the modeler. Each module describes a physical plant component to the prescribed level of fidelity and is independent of any other module. A module consists of a lumped parameter model, derived from first principles, describing a physical process by means of a system of non-linear algebraic and/or differential equations. A single component can be represented by different modules of different level of complexity to meet different modelling needs.

The basic characteristics of the LEGO package, can be summarised as follows:
- modularity: component models (modules) are available for general plant components (such as pipes, valves, pumps, heat exchangers, tanks, etc.) and the user can connect them in accordance with a specific plant design;
- flexibility: the user can solve special modelling problems by developing the mathematical model of special components, which can be included in the module library of the package;
- reliability: all the numerical algorithms used by the package are centralised in the master program. Module notifications, due to different mathematical modelling assumptions, do not require any numerical algorithm updating.
Moreover the modules can exchange information among themselves only by means of the master program so that they can be considered independent.

With respect to the numerical problem the main features of LEGO are:
- simultaneous solution of all non-linear algebraic and differential equations, using an implicit formula for the numerical integration method.
- use of sparse matrix techniques in order to reduce computation time, dealing with large power plant models.
- steady-state computation, allowing interchange of the role of input variables, output variables and uncertain constant parameters.

2. Sampierdarena Cogeneration Power Plant Model

The Sampierdarena power station is a combined cycle plant dedicated to produce both electrical and heat power. The thermal power is sent to final users as superheated water. The main plant components are: 1 gas turbine rated 21 MW, 1 steam generator where the residual heat of turbine exhaust gases is recovered, 1 steam turbine rated 9 MW, 1 generator rated 37.5 MWe, 1 583 m³ steam condenser, 1 deaerator. The gas turbine, the electric power generator and the steam turbine are placed on the ground mounted on the same axis as a unique group. The reference plant has been modelled by means of LEGO code modelling tools. The resulting simulation model is made up of 8 submodels: four of them reproduce process subsystems and the other ones reproduce automation system. More than 2000 variables are handled.

3. Diagnostic System Concepts

The theory of this diagnostic system is related with the usage of learning methods for automatic generation of knowledge bases. This approach enables the conversion of "cause→effect" relations into "effect→possible causes" ones. As applied to technical diagnosis, this allows one to generate troubleshooting rules of the kind "symptoms→possible diagnosis" through "failure→symptoms" relations. The problem of diagnostic rule generation is, thus, reduced to obtaining samples of symptoms, that is the well known problem of failure modelling. The following approaches to this problem may be used, in particular:
- failure modelling by experts;
- qualitative failure modelling technologies;
- computer process simulating programs;
- data of real physical failure modelling.
With the expert approach to modelling, the process experts are asked to describe manifestation of particular failures rather than to formulate diagnostic rules. To this end, the experts by means of special software tools can input interactively the descriptions of symptoms. The resulting sample of descriptions is processed by a special program relying on the algorithm identifying groups of characteristic features. As a result, diagnostic rule modules are generated automatically.

![Diagram](image)

*Figure 1. Functional architecture of the prototype system*

### 4. Diagnostic System Implementation

The system relies upon a teaching procedure enabling automatic generation of large diagnostic knowledge bases. The basic idea is to create diagnostic skill functions based on a wide experience of plant response following a number, as large as needed, of initiating events and/or system and component malfunctions.

The technical implementation of the concept is constituted by three main steps:

- The first step is the identification of all initiating events and the corresponding operational modes.
The second step is to perform simulation of all selected transients for a time sufficient to identify the uniqueness of the plant response under each initiating event.

The third step is to automatically generate a set of diagnostic rules that, through the use of temporal logic and a filtering Expert System, associate the evolution of the process parameters and their derivatives to the initial perturbation.

The following types of failures can be detected:

- leakages;
- cut off or deterioration of the characteristics of flow/pressure plant components;
- pipeline fluid conductance disturbances (valve defects, defects of controllers, foreign objects, defects of pipeline process equipment such as filters, heat exchangers, etc.);
- excessive passage of the medium through pipelines (valve defects, defects of controllers);
- defects of control systems of auxiliary mechanisms;
- sensor defects;
- damage of equipment elements.

The identification of malfunctions is performed through the use of the fuzzy set logic algorithms.

5. Functional architecture of the prototype

The conceptual flow diagram of the IDA prototype application is shown in fig. 1. The current information on process variables should be obtained in general from the standard monitoring system sensors. In the prototype it comes from the plant simulator and is fed into the filtering expert system generating a qualitative description of the process time profile. The hardware configuration of the prototype system is made up of a network of a HP workstation and a Digital VAX-Station. The IDA prototype runs on the HP workstation and the plant simulator runs on the VAXStation. The filtering Expert System is a software shell that is adjusted to a particular process plant using the data from the following sources:

- technical documentation of the process plant;
- the result of numerical simulation of various operational modes;
- calculations of process experts.

The following information is input into the Expert System:

- information about plant operating modes;
- the limits of process variables variations in static and transient modes.

The qualitative description of the process is sent to the Diagnostic Knowledge Base where a task is running that handles qualitative failure diagnosis rules built in into it. The diagnoses generated here include location, time, type and confidence of failures. They are fed into the Interpreting Expert System that executes another task and sends the diagnoses with assigned sources into the Graphic User Interface. The Graphic User

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Interface displays the received diagnoses on the Hewlett-Packard workstation console and fixes them to particular points in the mimics.

6. Rationale for the Early Diagnostic technology development

The Human-Machine Interface (HMI) technologies, as a mean for the most effective representation to the operator of the plant condition, and the consequent selection of the most appropriate recovery procedure, rely upon two basic approaches: (i) the "cognitive" approach and (ii) the "skill based approach".

6.1 The knowledge-based approach

In a “knowledge-based” (or cognitive) approach, a set of plant functional goals are established, based on the analysis of the operations and the overall aims of the control staff. This method is particularly strong towards unfamiliar situations, faced with an environment for which no know-how or rules for control are available from experience (for instance, multiple and concurrent failures). The control of plant performance is at a very high conceptual level, in which performance is goal-controlled. The decision-making process is established on the basis of the understanding of the functional properties of the process and prediction of the effects of the plant considered. The plant is represented through a functional model, that can assume different forms.

The information necessary for decision-making must be perceived in terms of symbols, i.e., concepts tied to the functional properties of the plant and represented in a way as to reinforce this functional perspective.

The HMI Systems that follow the knowledge based approach must communicate to the operator in terms of high abstraction level concepts, such as mass and energy balances, and must make clear, at any time, the reasons why a given control action is to be accomplished.

This approach requires a great deal of analysis of the plant functions, to find out all of the potential goals and their implications.

6.2 The skill-based approach

In the “skill-based” approach, the HMI Systems design exploits the already established sets of rules or procedures that have been derived empirically during previous occasions, or analysed through ad hoc evaluations. Hence, it is mainly founded on a consolidated background of experience of plant operations through hundreds years of reactor operation.
Though this approach is goal-oriented, the goals are not explicitly formulated, but are found implicitly in the situation releasing the stored rules. The control of the plant and of its systems is selected from previous successful experiences, and evolves by the survival of the fittest rule. This latter will reflect the functional properties of systems and components. With this approach, the goals will only be reached after a long sequence of acts, and direct feedback correction considering the goal is not possible, unless resorting to a functional understanding of current response of the plant, i.e., to a knowledge-based model.

The plant is seen in terms of aggregate of hardware (systems), with functional properties. The type of information presented to the operator with this approach relates to the performance of plant systems; for example, it emphasises the present behaviour of a given system with what the past experience suggests to be the optimum behaviour of that system.

The operator's decision is then made in terms of rules that dictate the strategy to restore the full functionality of that system.

The HMI Systems that follows the rule-based approach must communicate to the operator in terms of systems and components parameters. It is task of the operators then, to integrate this data into more useful aggregates.

This approach requires a great number of analysis of accident conditions to establish an appropriate set of plant procedures.

6.3 The goals of IDA

Both the approaches above summarised become effective only following the detection of a macroscopic fault and/or the violation of a Safety margin, that is, when the plant conditions are sufficiently far from the normality ones to allow, with good confidence, the identification of the recovery procedure.

The aim of IDA is to prevent the accident evolution by a timely trouble shooting process during all plant operational modes. It means that any deviation of plant parameters from normal values (at any operational mode) is pointed-out to plant operators well before the reaching of any undesired threshold potentially leading to not allowed plant state, together with the cause that has generated the deviation.

IDA is an additional system to existing already developed HMI Systems like safety parameters presentation system, operator adviser system, etc. which covers early period of faults development appearance.

Existence of such an additional system allows:

a) alert the operators sufficiently early to identify a beginning of possible fault and consequently to eliminate it;
b) move exactly to identify possible initial event of accident which decrease probability of creation of initial event of accident.

7. Validation of the Diagnostic Rules

The knowledge base built-up with appropriate tools in accordance to Sections 3 through 5, has been tested extensively. The goal of this plan was to evaluate the global performance of the application while running coupled to the simulator used for the learning phase. The following list of initiating events was used to drive the simulator, during independent transients:

a) pipe leaks (2);
b) pump abnormal stops (4);
c) miscellaneous valve malfunctions (35);

A transient duration of about 600 s after the initiating event was choosen. The actual testing methodology was based on the analysis of the recorded output of each transient. The content of the output is just a trace of the system dynamic performance, i.e. a sequence of diagnostic hypothesis with an associated certainty factor, delivered at specific times. A diagnostic hypothesis can include one or more alternative diagnostic messages. A diagnostic message is a reference to a specific initiating event. As an obvious assumption, a diagnostic hypothesis is acceptable if it contains a reference to the current initiating event, otherwise it is unacceptable. Between two distinct acceptable hypotheses the better one has a higher certainty factor, or a lower number of diagnostic messages when certainty factors are equal. Taking into account the previous concepts, the following parameters have been identified and recorded:

- \( N_h \): the total number of hypotheses delivered after the current initiating event;
- \( N_a \): the number of acceptable hypotheses;
- \( A \): the set of the acceptable hypotheses;
- \( C_k \): the certainty factor of the \( k \)-th hypothesis;
- \( N_m \): the number of messages contained in the best hypothesis;
- \( t_a \): the time in seconds after the introduction of the initiating event when the first acceptable hypothesis is delivered;
- \( t_b \): the time in seconds after the introduction of the initiating event when the best hypothesis is delivered;
- \( t_w \): the time in seconds after the introduction of the initiating event when the best hypothesis is withdrawn.

To the end of carrying out a quantitative evaluation on the base of previous parameters, some “properties” have been defined, with an associated conventional scoring system, ranging 0 through 10. These properties are listed herebelow, together with their definitions:

a) certainty, \( C \):
b) promptness, \( P \):

\[
C = \frac{10 \sum_{a \in A} C_a}{N_a}
\]

\[
\begin{array}{|c|c|}
\hline
\text{if} & P \\
\hline
t_a > 600 & 0 \\
600 > t_a > 550 & 1 \\
550 > t_a > 500 & 2 \\
500 > t_a > 450 & 3 \\
450 > t_a > 400 & 4 \\
400 > t_a > 350 & 5 \\
350 > t_a > 300 & 6 \\
300 > t_a > 250 & 7 \\
250 > t_a > 200 & 8 \\
200 > t_a & 9 \\
100 > t_a & 10 \\
\hline
\end{array}
\]

c) reliability, \( R \):

\[
R = \frac{10 \sum_{a \in A} C_a}{N_t}
\]

\[
R = 0, \text{if } A = \emptyset
\]

d) resolution, \( S \):

\[
S = \frac{10}{N_b}
\]

e) persistency, \( Q \):

\[
Q = \frac{t_{br} - t_{bi}}{50}
\]

With the previous definitions the results of the testing can be easily summarized in the tables 1-2.

**Table 1. Resulting scores of test (1)**

<table>
<thead>
<tr>
<th>Event</th>
<th>( N_a )</th>
<th>( N_s )</th>
<th>( R )</th>
<th>( C )</th>
<th>( P )</th>
<th>( S )</th>
<th>( Q )</th>
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<td>10</td>
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Table 2. Resulting scores of test (2)

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<th>( N_x )</th>
<th>( R )</th>
<th>( C )</th>
<th>( P )</th>
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520
To properly evaluate these results it is worth to consider that this is just a test of a demonstrative application. In addition, the score system proposed is a first approximation that should be refined in order to represent the differences in a smoother way.

8. An application to a VVER-1000 plant simulator

An application of this technology to a VVER-1000 plant is actually under way. ASTRA, an engineering simulator for this plant, has been developed in Ansaldo. The following subsystems have been modelled up to now:

- Reactor Coolant System and Reactor Core;
- Steam Generators and Main Steam System;

The following subsystems are going to be modelled:

- Makeup-blowdown system;
- Oil system for makeup pumps;
- Cooling water for makeup and oil system heat exchangers;
- Pressurizer System (including the relief tank);
- Simplified Turbine Cycle;
- Main Associated Control Systems (continuous and discrete).

The implemented model shall allow to simulate and then build rules to diagnose specific disturbances:

- deterioration of operation or shutdown of a source (pump, gas supply, ejection devices, receiver, etc.);
- deterioration of pipeline fluid conductance due to clogging;
- deterioration of valve fluid conductance due to mechanical damage or noncorrespondence of the valve position to the command of operator or control system;
- deterioration of heat exchanger fluid conductance due to mechanical damage;
- deterioration of filter fluid conductance due to mechanical damage or changes in the characteristics of filtering elements (exhaustion of filtering capability, caking, etc.);
- excessive passage of the medium through the valve due to mechanical damage or noncorrespondence of the valve position to the command of operator or control system;
- defect of the control system;
- defect of data sources (sensors);
- leaks of pipelines and equipment to the environment through the pressure boundaries;
- leaks through the gate of a closed valve.
9. Conclusions

As a matter of fact we can observe that many expert systems developers face the problem of building a consistent knowledge base. Experts are very often not available or, if available, they offer their collaboration as spots, with communication problems (the language of the expert is different from the language of the analyst). This fact increases the difficulties encountered during the application development and maintenance.

A better practice should be to avoid problems at the source, switching the source of the knowledge from the expert to the designer. Is this possible? This paper is an attempt to give an answer to this question.

The solution proposed is a computerised system based essentially on a simulator able to provide the plant behaviour and an expert system that supplies the knowledge to detect malfunctions.

The elicitation of this knowledge is acquired in a semi-automatic mode, introducing a number of malfunctions while the simulator runs. Malfunctions are introduced one at a time, and after every malfunction is introduced, the system transient is analysed to extract rules that describe the consequences.

The diagnosis of possible malfunctions is then made by a module that monitors data coming from the plant, or the simulator, and uses the rules generated in the elicitation phase, taking into account possible overlaps between premises common to more than one conclusion. If more than one conclusion can be drawn, for each one is supplied an evaluation of its degree of confidence.

As a brief statement to conclude: the practical experience done has enforced the initial assumption that automatic rule generation is possible and profitable. The knowledge elicitation methodology is simple and quick. New malfunctions can be introduced into the simulator and the new rules generated can be added to the previous knowledge base, and the knowledge base integrity and coherence can be automatically verified to assure always appropriate diagnosis.

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SIMULATION STRATEGY
FOR MANAGING NUCLEAR EMERGENCIES
AT JOSE CABRERA NUCLEAR POWER PLANT

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J. Blanco, J. Saiz, E. Moralo,
P. Almeida, C. Gómez

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**Simulation Topics**

- Technical management of NPP requires:
  - High safety levels
  - Balanced output in a competitive market

- To attain these goals is necessary:
  - In depth knowledge of the process
  - An optimized operation

- Regulatory authorities demand safety guarantees even in beyond design basis accidents (severe accidents)

- Preventive measures and mitigating actions are needed

- Simulation allows evaluation of the effectiveness of these procedures

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**Simulation Topics**

- Transient accidents severe acc. scenarios
  - Training
  - Operation
  - Procedure optimization

- Benefit optimization
- Fuel management
- Engineering
- Safety analysis

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**SIMULATION TOPICS**

- JOSE CABRERA NPP HAS SINGULAR DESIGN FEATURES
- GENERICAL ANSWERS FOR SEVERAL TOPICS (E.G. SAFETY ISSUES) ARE NOT ALWAYS APPLICABLE
- TAILORED SOLUTIONS MUST BE MADE BASED ON BEST PLANT KNOWLEDGE
- SIMULATION PLAYS A MAJOR ROLE IN JUSTIFYING PARTICULAR ANSWERS AND FINDING OPTIMAL SOLUTIONS
- QUALIFIED TOOLS ARE REQUIRED FOR A WIDE RANGE OF SCENARIOS
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SIMULATION CAPABILITIES AT JOSÉ CABRERA NPP

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JOSÉ CABRERA TECHNICAL STAFF HAS PARTICIPATED IN SEVERAL INTERNATIONAL PROJECTS:

- OECD-LOFT
- LOBI
- ICAP
- CAMP
- PHOBUS-CSD
- LACE
- ACE

THERE WAS A TWOFOLD PURPOSE:

- STATE-OF-THE-ART CODE ACQUISITION
- CODE USAGE IN SPECIALIST ENVIRONMENTS

- INTRODUCTION
- SIMULATION TOPICS
- JOSÉ CABRERA NPP SIMULATION CAPABILITIES
- NUCLEAR EMERGENCIES MANAGEMENT AT JOSÉ CABRERA NPP
- CONCLUSIONS
**NUCLEAR EMERGENCIES MANAGEMENT**

- **IN CASE OF AN EMERGENCY, CONTROL ROOM OPERATORS AND TECHNICAL SUPPORT CENTER (TSC) STAFF DIRECT THEIR ACTIONS TO AVOID OR MITIGATE CORE DAMAGE AND TO MINIMIZE THE IMPACT ON PLANT SYSTEMS AND THE ENVIRONMENT**

- **SEVERAL TOOLS HAVE BEEN (OR ARE BEING) DEVELOPED TO:**
  - OPTIMIZE THEIR ACTIONS
  - DECISION-MAKING HELP
  - CONTINUOUS TRAINING

---

**NUCLEAR EMERGENCIES SIMULATION**

- EOP
- SGIZ
- PLANT MODELS
- NPA-Z
- SAGAZ
- SAGAS-Z

**EMERGENCY OPERATING PROCEDURES**

**TRAINING SIMULATOR**

**DIFFERENT CODES**

**B-E TRANSIENT ANALYZER**

**EMERGENCY MANAGEMENT DBA**

**EMERGENCY MANAGEMENT SEVERE ACCIDENTS**

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NUCLEAR EMERGENCIES MANAGEMENT

- THE EMERGENCY OPERATING PROCEDURES (EOP) WERE DEVELOPED SPECIFICALLY FOR THE PLANT

- THEY HAVE BEEN EXTENSIVELY VALIDATED IN SEVERAL WAYS:
  - ANALYTICAL
  - PLANT SIMULATORS
  - EXPERIMENTS (LOBI BL-40 FOR SGTR)

- THE SGIZ SIMULATOR WAS MADE SPECIFICALLY FOR JOSÉ CABRERA NPP.

- MAIN FEATURES:
  - REAL-TIME
  - BASED ON ADVANCED SIMULATION MODELS (TRAC-G)
  - SUITABLE FOR TRANSIENT AND ACCIDENT SITUATIONS PLANT SIMULATORS
  - APPROPRIATE SCOPE FOR OPERATORS AND TSC TRAINING IN EOPs

NUCLEAR EMERGENCIES MANAGEMENT

- SEVERAL REALISTIC MODELS HAVE BEEN BUILT FOR DETAILED T/H ACCIDENT ANALYSIS

- THEY HAVE BEEN VALIDATED IN DIFFERENT WAYS:
  - FORMAL INDEPENDENT REVIEWS
  - EXERCISES IN INTERNATIONAL ASSESSMENT PROJECTS (LOFT, ICAP, CAMP)
  - DETAILED COMPARISONS WITH ACTUAL PLANT TRANSIENTS

- COMPLETED MODELS FOR JOSÉ CABRERA NPP:
  - COBRAIII/MIT
  - TRAC-PFI/MOD1
  - RELAPS/MOD2
  - RELAPS/MOD3
  - MAAP3.0B
  - MAAP4
  - SCDAP
NUCLEAR EMERGENCIES MANAGEMENT

- BASED ON RELAPS. MAIN FEATURES:
  - SCOPE: NSSS
  - INTERACTIVE CAPABILITY
  - BEST-ESTIMATE RESULTS
  - GRAPHICAL PRESENTATIONS ADAPTED FOR ACCIDENT SIMULATION
  - POSSIBILITY OF SIMULATE THE PRINCIPAL EOP STEPS

- SLOWER-THEAN-REAL-TIME
• S.A.G.A.Z. MEANS ACCIDENT MANAGEMENT SUPPORT SYSTEM FOR JOSÉ CABRERA NPP.
• IN SPANISH, SAGAZ MEANS WISE
• MAIN FEATURES:
  • ON-LINE CONNECTION
  • CAPABILITY TO PROCESS AND ANALYZE PLANT DATA
  • DIAGNOSE SITUATIONS
  • NON-SEVERE ACCIDENTS
  • FASTER-THAN-REAL-TIME SIMULATION
  • INCLUDES THE MAJOR EOPs
  • MAAP4 BASED
• STATUS: DEVELOPMENT PHASE
### Nuclear Emergencies Management

- **Integral Code (NSSS, Containment, Auxiliary Buildings and Systems)**
- **Transient and Accident Simulation**
- **Easy to Simulate Operator Action**
- **Interactive Graphical Interface**
- **Faster-Than-Real-Time**
- **Portability**
- **Usage Experience (PSA Engineering Support Analyses, Add-Ons)**
- **Plant Model Validation Versus Other Codes**

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**SAGAS-Z** means Severe Accident Management Support System for José Cabrera NPP.

- **Functions:**
  - Severe Accident Simulation and Operator Training under the Severe Accident Management Guidelines (S/AMG)
  - Providing Predictive Capacities Using MAAP4 Code and Fuzzy Logic Techniques.
- **Status:** Initial phase.
- **Currently,** the works will start within a collaboration in CAMS area of Halden project.
INTRODUCTION

SIMULATION TOPICS

JOSÉ CABRERA NPP SIMULATION CAPABILITIES

NUCLEAR EMERGENCIES MANAGEMENT AT JOSÉ CABRERA NPP

CONCLUSIONS

CONCLUSIONS

- IT HAS BEEN PRESENTED THE SIMULATION STRATEGY OF JOSÉ CABRERA NPP FOR MANAGING NUCLEAR EMERGENCIES.

- SIMULATION STRATEGY HAS HELPED TO FIND BETTER SOLUTIONS FOR MANAGING NUCLEAR EMERGENCIES AT JOSÉ CABRERA NPP.

- THERE IS A SOUND EXPERIENCE IN USING T/H SIMULATION CODES, WHICH NOW MAKES POSSIBLE THE DEVELOPMENT OF GENERAL PURPOSE TOOLS BASED ON THEM.

- COMPUTER TOOLS TO HELP IN NUCLEAR EMERGENCIES MANAGEMENT IS NOW UNDER DEVELOPMENT.

- JOSÉ CABRERA NPP OPERATORS AND TSC TRAINING WILL RANGE FROM OPERATIONAL TRANSIENTS TO SEVERE ACCIDENT MANAGEMENT.

- FUTURE WORKS ARE FORESEEN, INCLUDING THE MOST IMPORTANT EMERGENCY PROCEDURES AUTOMATIZATION AND A COMMON GRAPHICAL INTERFACE.
EXPERIENCE OF APROS IN NUCLEAR POWER PLANT SAFETY ANALYSES

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Abstract

In 1995 Imatran Voima Oy (IVO) decided to implement a modernization and power uprating program for Loviisa Nuclear Power Plant (NPP). Among other things Loviisa Final Safety Analysis Report (FSAR) had to be extensively revised as part of the licensing process for higher reactor thermal power. In 1995-97 the major part of FSAR safety analyses were recalculated using the uprated 1500 MW as a nominal power. Main tool in the analysis work was a simulation software called APROS. This paper gives an overview of the APROS analyses relating to the revision of Loviisa FSAR.

1. Introduction

In the Loviisa modernization and power uprating project major part of the revised Loviisa FSAR thermal hydraulic analyses were calculated using the APROS code. Only a few initiating events were analyzed with other codes. Events in which asymmetric neutron kinetics played an important role were analyzed using the HEXTRAN-SMABRE code. Moreover, containment maximum and minimum pressure analyses were calculated with the RALOC code.

This paper gives first a short overview of the Loviisa modernization and power uprating project and then a few words are written about the APROS code itself. This is followed by the discussion of the development and validation of the Loviisa plant APROS input model. In the following section initiating events which were analyzed using APROS are listed and after that, to demonstrate APROS simulation capability, a short discussion takes place of three initiating events which were considered to cover several important phenomena which a safety analysis simulation code must be able to calculate. Finally
based on the whole analysis work a discussion of the experience gained from the APROS code takes place.

2. Loviisa modernization and power uprating project

A feasibility study for modernizing the two 465 MWe VVER units of Loviisa NPP was carried out starting in spring 1994. During the study no technical, safety or licensing issues were identified which would have prevented raising the reactor thermal output up to 1500 MW from the level of 1375 MW. Thus the modernization project including a 9.1 % reactor power uprating was launched in summer 1995.

It was obvious from the very beginning that reactor power uprating would bring about a need for extensive revision of the Loviisa FSAR including the safety analyses. It was also clear that tasks related to the safety assessment by appropriate computer codes and models would be on the critical path on the project schedule. That is why this work was started at the same time as the project organization was put together and the master plan prepared, roughly two years ago. TVO aims to receive a license for operation of the reactors at 1500 MWth at the beginning of 1998.

The increase in the plant capacity, by about 50 MWe per unit, is achieved by a combination of reactor thermal power uprating and by improving turbine efficiency. Changes in the main process parameters connected to the 9.1 % reactor power uprating are presented in Figure 1. The temperature difference between the primary coolant entering the reactor and the coolant at the outlet will increase in proportion to the increase in thermal power. Thus, the rise in the average temperature difference over the reactor core will be slightly less than 3 °C. The primary coolant flow rate and the primary circuit pressure will remain unchangeable.

The live steam flow rate to the high pressure (HP) turbine will increase in proportion to the thermal power, but the steam pressure is not affected. Regarding the service water system, the flow rate will remain the same resulting to a temperature rise of about 1 °C in the condenser discharge flow back to the sea.

As to the core fuel load, the present limits for the maximum linear power and burn up will remain. The rise in thermal output will be achieved by reducing the power distribution peaking factors and increasing the number of fresh fuel assemblies in the core, in expense of the fuel economy. In parallel, other possibilities for increasing the thermal margins and for optimizing fuel costs in the long term have been investigated, such as effects of increasing fuel enrichment.
The revision of transient and accident analyses now completed, included not only computer calculations of selected cases but also constructing a totally new model for Loviisa Unit 1. APROS was chosen to be used for the majority of the thermal hydraulic analyses, and the new model was extensively validated against measured plant data from commissioning tests and against results of previous licensing analyses.

3. Applied analysis tool APROS

IVO Power Engineering Ltd and the Technical Research Centre of Finland have developed APROS Simulation Software since 1986. APROS [1] is a multifunctional simulator, which is used for process and automation design, safety analysis and training simulator applications. APROS has in these applications many useful and also unique features:

- Physical and accurate dynamical models
- Graphical, interactive and easy modelling
- Fast running simulator
- Computer independence
- Whole plant including the process, automation and electrical systems can be modelled and the connection of these to different systems is easy to do
- Design model can be used as a training simulator by adding appropriate displays

The performance of the APROS code has been validated extensively in more than 60 cases by modelling different test facilities and comparing calculations to a large set of selected transients [2].

4. Scope of the analysis work

The analysis work was carried out in the following steps:

- Collection of data needed for the input model and construction of the Loviisa Unit 1 APROS input model
- Validation of the input model
- Calculation of the chosen initiating events

4.1 APROS model of Loviisa NPP

There are two units at the Loviisa site of which Unit 1 was chosen to be modeled. Basically both units are similar but there are some differences like higher (approx. 5 %) primary coolant mass flow rate in Unit 2.
For the Loviisa FSAR analyses a completely new APROS model was developed from scratch. The developed model included a detailed description of the primary circuit with all six loops modeled separately. Emergency core cooling systems (ECCS), make-up, letdown and also purification systems were modeled including water tanks, pipe lines and pumps.

The secondary side included a detailed model of each steam line separately from steam generators to the turbines. Also all the main and emergency feed water pipe lines and pumps were modeled in detail from feed water tanks to the steam generators.

All the main control systems were included in the model

- Primary circuit pressure control
- Pressurizer level control
- Reactor power control (secondary side pressure, neutron flux, number of operating reactor coolant pumps)
- Turbine power control (number of operating reactor coolant pumps, number of operating main feed water pumps, manual set point, power control for each turbine separately)
- Steam generator level control
- Control system for loss-of-electricity and diesel generators start-up sequence

Also all the protection systems and signals including

- Reactor protection
- Plant protection
- Brittle fracture protection

were included in the APROS model.

4.2 APROS model validation

It is of great importance to validate the completed model in order to prove that the model geometry corresponds to the real one, all the signals are included and the actuation set points are correct, pumps and valves are modeled with correct capacities etc.

The first stage of the APROS Loviisa model validation was to calculate the steady-state situation on different power levels (100 % = 1375 MW, 109 % = 1500 MW, 111 % = 1530 MW) and to compare with measured or precalculated ones. Then a natural circulation rate on various power level (0.5, 1.0, 1.5, 2.0, 2.5, 3.0, 3.5 and 4.0 %) was compared with the data which was previously produced from plant measurements. After that selected plant commissioning tests were calculated and the results were compared with
the measured data. These tests included reactor coolant pump trips (one pump and six pumps) from 100 % power level, reactor trip from 100 % power level and trip of one turbine from 96.7 % power level. Also a stuck open turbine by-pass transient which took place in Loviisa 2 in 1981 was calculated and results were compared with measured plant data.

The validation program was continued by repeating the cases calculated previously with RELAP5. These analyses included small break loss-of-coolant accident (SBLOCA), primary-to-secondary side leakage (PRISE) accident and a HEXTRAN analysis concerning loss of on and off-site AC power as an anticipated transient without scram (ATWS) case. Also large break loss-of-coolant accident (LBLOCA) was calculated with APROS and the results were compared with the earlier DRUFAN/FLUT analysis results.

Final validation concerned a reflood experiment performed in REWET test facility. This made an important contribution for assessing the capability of APROS to calculate LBLOCA and particularly its reflood phase.

4.3 Initiating events analyzed using APROS

As already shortly discussed in Chapter 2 the revision of the Loviisa FSAR the new uprated 1500 MW thermal power required a great number of safety analyses. The initiating events which were considered to be the limiting cases were analyzed in the first phase. Later on all the initiating events presented in Loviisa FSAR will be updated to correspond to the uprated power level.

APROSO code was chosen as a main tool in this work. The choice was based on facts like during the code development work special Loviisa reactor features have been included to the code and assessment with the PACTEL facility has been done extensively. APROS has also gone through a wide validation program and it has successfully been used in process and automation design for Loviisa NPP.

6-equation version of the APROS code was used in all the analyses. The initiating events which were analyzed using APROS code are listed below:

- LBLOCA (Cold and hot leg, BOC and EOC)
- SBLOCA (Cold and hot leg, several break sizes)
- ATWS (CRW from full power, Loss of main FW, Loss of on and off-site AC power)
- PRISE (Single tube, medium, large)
- Reactor coolant pump trips (One, Three)
- Reactor coolant pump seizure
- Main feed water pumps trip
- Feed water line break (Three different break locations)
- Inadvertent closure of main steam line isolation valve (One valve, Six valves)
- Loss of on-site and off-site AC power
- Uncontrolled withdrawal of a control rod group during power operation
- Overpressure protection analysis (Six or one MSIV closure, Turbine trip)
- Decrease of feed water temperature
- Inadvertent opening of one steam generator safety valve

To study the sensitivity of the results, each initiating event analysis included several parameter variation analyses. Parameter variations included, e.g., in pipe break cases, variations of break size and break location, in ATWS and overpressure protection analysis, variation of initiating events and in analyses dealing with pump trips, variation of number of pumps to trip and power control system (ROM) operating or not. Also, capacity of ECCS and emergency feed water (EFW) pumps were considered important parameters as well as the availability of on and off-site AC power.

5. Results of selected analyses

Three initiating events were selected for closer presentation in this paper. Two events deal with primary circuit leakages and one is an ATWS case with an uncontrolled withdrawal of a control rod group as an initiating event. These events cover several important phenomena (blowdown, reflood, two-phase flow, core uncover, core power behavior, circumstances in high and low pressure) that a simulation code must be able to predict when it is applied to licensing type analyses.

5.1 Large break LOCA

LBLOCA is still considered one of the most difficult cases for simulation codes to calculate. This is due to the fact that phenomena especially during reflood phase are mostly three dimensional but the codes including APROS are usually designed for one-dimensional calculation. In the case of Lovisa, the situation becomes even more difficult because of combined injection configuration during hydro accumulator and low pressure emergency core cooling pump injection phase.

The initial power level in the analysis was 1530 MW (102%). The assumption concerning ECC systems was that one out of four high pressure and one out of four low pressure emergency core cooling pumps were available. Moreover, two out of four emergency accumulators, one injecting into upper plenum and the other into downcomer, were also available. The break was assumed to locate between reactor pressure vessel and reactor coolant pump in the cold leg case and in the hot leg case near the pressurizer discharge line connection point.
Since the reactor core consists of fuel assemblies with different power level it was considered important to make the APROS model closer to the reality than the earlier (late 1980's) LBLOCA core models. Earlier limitation in core modeling was computer capacity which did not allow very great number of nodes. Nowadays the situation is different, since the computer capacity and speed has increased tremendously during the past ten years.

A more realistic modeling was done in such a way that fuel assemblies were divided into four different groups based on fuel burnup. UO₂ thermal conductivity and gap conductance were then calculated for each group taking into account the fuel assembly or fuel rod power level. In the APROS model the core was divided into 7 radial channels and each channel into 40 hydraulic volumes in the axial direction. Fuel rods and assemblies were divided into 10 stacked heat structures inside each hydraulic volume. Three radial channels represented hot assemblies of 1., 2. and 3. cycle and hot rod of each cycle was put into the corresponding hot channel.

In the first phase of cold leg break only blowdown phase was calculated assuming different break sizes from the reactor pressure vessel side. Maximum break size was found to give highest peak cladding temperature. Then the whole accident was calculated from the beginning up to the point of time when the whole core had quenched assuming beginning of cycle (BOC) situation. The calculation was then repeated with the assumption of end of cycle (EOC) situation. The APROS analysis clearly indicated that LBLOCA combined with BOC situation results in higher peak cladding temperature (Figure 2) than EOC situation (Figure 3) during both blowdown and reflood phase. The main reason for higher peak cladding temperature (BOC) during blowdown phase was the higher stored energy in the core because of smaller gap conductance. The lower peak cladding temperature (EOC) during reflood phase was due to the fact that during the hydro accumulator injection phase hot assemblies in each hot channel quenched creating favorable cooling circumstances for each hot rod and thus preventing hot rod cladding temperature from rising anymore.

Hot leg LBLOCA analysis clearly showed as expected that core cooling is ensured during the whole accident and fuel cladding temperature followed the coolant saturation temperature all the time excluding a few short time small oscillations.

5.2 Small break LOCA

In Loviisa one of the main concerns in SBLOCA events is the rather wide pressure gap (1.2 MPa) between hydro accumulators being empty and shut off head of the low pressure injection (LPI) pumps. The only ECC system available on this pressure range is high pressure injection (HPI) system and in the worst case only one HPI pump is able to inject into primary circuit. If it takes a very long time until primary pressure has de-
creased below the LPI pumps shut off head and HPI pump is not able to compensate the break mass flow rate, there is a danger for the core uncovering and overheating.

In the SBLOCA analyses the following approach was used to find out if there is a break size which results in core overheating. The break location was chosen to be in the cold leg so that the injection of only one HPI pump was available. Then starting from the 40 cm² break area the analysis was repeated by increasing the break area to 80, 100, 140, 150, 160, 170, 180 and 200 cm². All other parameters remained unchanged.

The results showed that if the break area was < 100 cm² no major core overheating took place. When the break area was increased APROS predicted a clear core overheating for every break area between 100 - 170 cm² (Figure 4). Overheating continued from 150 s to 250 s lasting a longer period of time with higher peak cladding temperature. For break areas 180 cm² and 200 cm² the overheating rate was much smaller, lasted a shorter period of time and took place earlier.

The analyses revealed that the core overheating was clearly related to the point of time when the hydro accumulators emptied. After this primary circuit mass inventory and pressure vessel collapsed liquid level rapidly decreased and the upper part of the heated core uncovered (Figure 5). This was due to the fact that the downcomer level decreased below the cold leg elevation after the hydro accumulator injection stopped. Hence, also the collapsed liquid level in the upper plenum and core region dropped. The break flow changed from two phase flow to steam flow and after this the HPI pump was able to compensate the break flow and both primary coolant mass inventory and collapsed liquid level began to increase. This resulted in fuel cladding temperature degradation down to the coolant temperature level and after that cladding temperature followed the coolant temperature until the calculation was stopped at 2 h.

5.3 ATWS analysis

Anticipated Transients Without Scram accidents are not design basis accidents for Lovisa but the Finnish regulations require them to be analyzed. During the modernization and power uprating project three initiating events were analyzed

- Uncontrolled control rod group withdrawal from full 1530 MW (102 %) power
- Loss of on and off-site AC power
- Loss of main feed water

The first initiating event in the above list resulted in the “worst” situation and it will be discussed below in more detail.
The uncontrolled control rod group withdrawal from full 1530 MW (102 %) power began from the situation, where the control rod group located as deep in the core as allowed by the technical specification document. Electrical failure was assumed to cause the withdrawal.

As a result of control rod withdrawal reactor power increased reaching the maximum of 118 % and soon after that decreased down to 108 % level (Figure 6). When reactor coolant pumps stopped power began to decrease due to reactivity feedback from coolant temperature rise. At 800 s reactivity feedback had “killed” the fission power and reactor power was based on decay heat generation from that point of time onwards.

Primary circuit pressure followed the reactor power behavior in the early phase of the accident (Figure 7). When the reactor coolant pumps stopped causing primary coolant temperature rise also primary pressure began to rise. Neither pressurizer relief valve nor pressurizer safety valves were able stop the pressure rise and at 640 s a maximum pressure of 14.6 MPa was reached. Soon after that primary coolant temperature rise stopped and turned around which resulted in pressure degradation together with the open pressurizer safety valves. Later primary pressure reached a couple of times 13 MPa level.

Even though this event resulted in “worst” results primary pressure remained all the time well below the acceptance criterion (18.5 MPa). Also maximum fuel cladding temperature criterion was fulfilled since minimum DNB ratio was 2.03 which proves that fuel cladding temperature all the time followed primary coolant temperature.

6. Experiences of APROS analyses

Generally it can be said that APROS simulation code did a very good job in the Loviisa modernization and power uprating project. Several operational transients and postulated accidents were analyzed and not a single major problem was detected which would have made the user to think of changing over to another code. The 6-equation model combined with a very detailed plant model results in quite long calculation time also for operational transients. Based on the users’ experience the 5-equation model is often ten times faster and it can be reliably applied for the operational transients. 6-equation model is recommended to be used in LOCA analyses but general opinion of APROS users is that 5-equation model gives reliable results also in many postulated accidents excluding LBLOCA. One should also bear in mind that accuracy of the results and the computing speed do not usually go hand in hand. The accuracy of the model is often paid by the computing time.
APROS proved to be such an excellent tool on safety analysis field that IVO Power Engineering Ltd is in the future going to do basically all the safety analyses using APROS code. The earlier major tool RELAP5 code is not, however, going to be thrown into the trash can but it will have a role in assessing the APROS analysis results. In this way the safety analysis work will also have a new dimension in Finland; i.e. one code (APROS) is used mainly by the utility and the other (RELAP5) is a tool for the regulatory organization to evaluate the results of APROS analyses.

References


Acknowledgment

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Figure 1 Approximate changes in main process parameters
Figure 2  Hot rod cladding temperature behavior (1. cycle, BOC)

Figure 3  Hot rod cladding temperature behavior (1. cycle, EOC)
Figure 4  Fuel cladding temperature behavior vs. break area

Figure 5  Pressure vessel collapsed liquid level behavior vs. break area
Figure 6  Relative reactor power behavior

Figure 7  Primary circuit pressure behavior
3D Core Model for Simulation of Nuclear Power Plants: Simulation Requirements, Model Features, and Validation

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ABSTRACT

In 1994-1996, Thomson Training & Simulation (TT&S) carried out the D50 Project, which involved the design and construction of optimized replica simulators for one Dutch and three German Nuclear Power Plants.

It was recognized early on that the faithful reproduction of the Siemens reactor control and protection systems would impose extremely stringent demands on the simulation models, particularly the Core physics and the RCS thermohydraulics. The quality of the models, and their thorough validation, were thus essential.

The present paper describes the main features of the fully 3D Core model implemented by TT&S, and its extensive validation campaign, which was defined in extremely positive collaboration with the Customer and the Core Data suppliers.
1. Introduction

In December 1993 the German Training Center, Kraftwerks-Simulator-Gesellschaft (KSG), in association with, and on behalf of German and Dutch utilities, awarded Thomson Training & Simulation (TT&S) a contract for the design and construction of four optimized replica nuclear power plant simulators for the following power stations:

<table>
<thead>
<tr>
<th>Name</th>
<th>Owner</th>
<th>Site</th>
<th>Rated Power (MWe)</th>
</tr>
</thead>
<tbody>
<tr>
<td>Unterweser KKU</td>
<td>Preussen Elektra AG</td>
<td>Esenshann Niedersachsen Germany</td>
<td>1350</td>
</tr>
<tr>
<td>Neckarwestheim GKN</td>
<td>Gemeinschaftskernkraftwerk Neckar GmbH</td>
<td>Kirchheim Baden-Wuerttemberg Germany</td>
<td>840</td>
</tr>
<tr>
<td>Obrigheim KWO</td>
<td>Energie Versorgung Schwaben Badenwerk AG</td>
<td>Obrigheim Baden-Wuerttemberg Germany</td>
<td>357</td>
</tr>
<tr>
<td>Borssele KCB</td>
<td>N.V. Elektriciteits-Produktiemaatschappij Zuid-Nederland</td>
<td>Vlissingen Netherlands</td>
<td>480</td>
</tr>
</tbody>
</table>

All four power stations are of PWR type, and were built by Siemens AG of Germany.

The supply of the specific Core Data Package was awarded to the following independent suppliers:
- For KKU, GKN, and KCB: Siemens AG, Erlangen, Germany, in association with PreussenElektra AG, Hannover, Germany, for KKU.
- For KWO: Forschungsinstut fur Kerntechnik und Energiewandlung (IKE), Stuttgart, Germany.

It is well known that the simulation of Siemens-built power plants is particularly difficult, because of the completeness of the Control Room information and the plant control systems. These characteristics impose severe demands on the simulation models.

This paper describes the requirements which resulted concerning the Core model, the features of the 3D model implemented, and the procedures applied for model validation.
2. Core Model Requirements

The specified scope of simulation covers of course all operating conditions from cold shutdown to full power, including situations arising out of malfunctions and/or operator omissions or erroneous actions.

The main malfunctions directly affecting the performance of the Core model are mostly classical:
- control rod drops.
- faulty control rod insertion/withdrawal.
- control rod ejection.
- ATWS.
- LOCAs.
- loss of a reactor coolant pump (recovery without scram).

However, the specific design of the Siemens-built nuclear power plants imposes more stringent requirements on the simulation models. This is due to the three staggered systems which act on the reactor control rods, namely the control, limitation, and protection systems.

The purpose of the limitation system (Begrenzungssystem), which is particular to the Siemens design, is to monitor and limit the reactor power, in order to:
- enhance plant safety by guaranteeing that, in the event of protection actions, the initial reactor conditions shall be within a well-defined envelope of operating conditions.
- enhance plant economic performance by allowing, in well-defined incidental cases, a continued operation at lower reactor power, without scram. The frequency of actual reactor scrams is thus considerably reduced.

This limitation action is carried out basically by quick power reductions, by means of specially planned control rod drops. For example, in case of loss of a reactor coolant pump, instead of immediately scramming the reactor, the limitation system will drop several pairs of control rods, in a pre-programmed sequence, in order to quickly reduce reactor power to a well-defined lower level, within strict tolerances. The plant may therefore continue safe operation, and scram is thus avoided while respecting safety requirements.

Due to the narrow differences between the various thresholds for these different actions, the correct simulation of the detailed interplay between the three reactor control systems is very demanding on the performance of the Core model, the Reactor Coolant System (RCS) and Steam Generator models, and the Instrumentation & Control models.
It was therefore decided:
- to implement a fully 3D core model, in order to be able to simulate all the required phenomena in a totally physical way.
- to submit the Core model to an extensive validation campaign.

The Core model validation was to be carried out by comparison with:
- the results supplied by the design/fuel management codes which would be applied to generate the Core Data.
- the results from design and/or safety calculations.
- experimental plant data.

The Core model features, and its validation tests, are described in the following.

3. Core Model Characteristics

3.1 Introduction

In the last decade, TT&S has oriented its Core modeling policy in simulators towards achieving the closest possible match between the simulation model and the design codes which are applied for fuel management and/or design and safety calculations in the reference power plant.

This policy, whenever feasible, presents obvious advantages:
- the Core Data Package may be generated and implemented with a minimum effort of handling and tuning.
- the results of the simulation model may be readily assessed by direct comparison with the design code.
- future updates of the simulated core characteristics may be implemented with considerable less effort.

In the case of the D50 Simulators, it was established early on that the design codes applied by the independent Core Data suppliers (see §1. hereabove) did not generate the Core Data in the form which had been foreseen by TT&S during the contract award phase.
Two alternative solutions were then explored:
- the off-line generation of "correction terms", in order to bridge the gap between
design calculations and simulation model for the different operating conditions
- the implementation by TT&S of a simulation model which would directly use the
  Core Data as generated by the design codes.

The first solution was shown to be unquestionably easier to implement, which is not
really surprising. However, in view of the extensive scope of simulation and the high
performance level required, it was nonetheless decided to make a decided attempt at
implementing the second method, while proceeding simultaneously with the first one as a
"safety net".

This second way is of course much more elegant and satisfactory from a technical point
of view, but obviously it is also fraught with difficulties. Its implementation required in
fact several developments before it was possible to satisfy the simulation requirements,
while simultaneously holding the necessary computing power within acceptable limits.
Furthermore, an extensive feasibility study had to be conducted in order to validate this
solution.

These difficulties were however substantially overcome, thanks to a very active
collaboration between all parties involved, and all D50 simulators are currently being
used for plant personnel training.

### 3.2 Final Core Model Design

In order to satisfy the requirements set off in §2 hereabove and to apply the Core Data
as directly as possible, the final version of the Core model was a straightforward
implementation of the neutron diffusion equations.

Figure 1 shows an overview of the calculation scheme - which is quite standard - and the
main model features are described in the following sections. The time step is 0.125
seconds (8 Hz).
Figure 1: Core model overview
3.2.1 Core Physics

The model was developed along the now classic lines of separation of the neutron fluxes into shape and amplitude functions, which are well established in the references [1-3]. We therefore express the total 3D flux distributions as follows:

\[ \Phi(x, y, z, t) = S(x, y, z, t) \times T(t) \]

where \( S \) is the shape function, and \( T \) the amplitude.

In the case of real-time simulation, however, the notion of macro and micro time intervals, which apply to shape and to amplitude calculations respectively, is not exactly the same as in design or safety calculations.

In the simulator, in order to accommodate fast transients, core reactivity must necessarily be updated at least at every time step of 0.125 seconds. If 3D calculations are performed less frequently, then some pre-calculated technique must be applied in order to estimate reactivity updates during the gaps.

Our goal, however, was to derive core reactivity directly from the neutron diffusion equations, without using any ad-hoc transition technique. This led, inevitably, to adopting the simulation time step as the macro time interval, and introduced much uncertainty on the computer power which would be required by the model.

In the final model version, the 3D calculations effectively determine the flux shapes and the core reactivity at every simulation time step.

The main model characteristics are:

- the shape function and the core reactivity are determined by solving the complete steady-state 3D neutron diffusion equations with two neutron energy groups:

  Fast neutron group:

  \[ \nabla \cdot (D_1 \cdot \nabla \Phi_1) - (\Sigma_{a1} + \Sigma_s) \cdot \Phi_1 + \frac{1}{\lambda} \cdot (\nu \Sigma_{f1} \cdot \Phi_1 + \nu \Sigma_{f2} \cdot \Phi_2) = 0 \]

  Thermal neutron group:

  \[ \nabla \cdot (D_2 \cdot \nabla \Phi_2) - \Sigma_{a2} \cdot \Phi_2 + \Sigma_s \cdot \Phi_1 = 0 \]

  where nomenclature is standard:
  - \( \Phi_1, \Phi_2 \) : neutron fluxes.
- $D_1, D_2$ : neutron diffusion coefficients.
- $\Sigma_{a1}, \Sigma_{a2}$ : macroscopic absorption cross sections.
- $\nu \Sigma_f, \nu \Sigma_r$ : macroscopic neutron production cross sections.
- $\Sigma_r$ : macroscopic removal (moderation) cross section.

One 3D neutron diffusion calculation is performed at every time step.

The complete 3D neutron balance equations were applied for each core cell. The convergence of the 3D fluxes was accelerated by solving the one-dimensional axial equations in parallel, at a faster rate. The improved axial flux shape thus obtained was applied to predict an average neutron axial leakage, which was then re-injected on the 3D calculations.

The core parameters for the axial calculation are determined by flux-weighted averaging over each axial core layer, using the 3D flux distribution. The radial neutron leakages are also computed using the 3D fluxes.

Global core reactivity is obtained as the eigenvalue of the axial calculation.

The core nodalization is as follows:

- axially, the model was adapted to use as many active layers as the Core Data supplier (13 to 16, in the present cases), plus two reflector layers at top and bottom. The cell axial width is non-uniform, in order to use a simulation mesh identical to that of the design codes; this adaptation greatly simplified data handling and implementation.
- radially, one core cell (or node) per fuel assembly.

There are therefore a total of about 3700 3D core cells (or nodes) in the case of the larger Unterweser (KKU) core, including the reflector.

This nodalization allows of course the direct individual simulation of incore nuclear detectors and core outlet thermocouples, without any need for local flux reconstruction techniques or empirical weighting coefficients.

- the reflector is included in the radial and axial calculations.
- the xenon/samarium concentrations are dynamically calculated for each individual core cell.
- the decay heat power density is dynamically calculated for each individual core cell, using 11 groups of fission product groups.
the amplitude function is calculated by integrating the kinetic equations, using 6
groups of delayed neutron precursors.

The core physics parameters (i.e., diffusion coefficients and cross sections) were
expressed as polynomial functions of the thermodynamic variables and boron
concentration. The real-time polynomials were fitted by the method of least-squares for
each 3D core cell.

In addition to requiring precise approximations for these functions, the accurate
calculation of core parameters depends of course on achieving detailed and correct
values of the thermodynamic conditions. This requirement is addressed in the following.

3.2.2 Detailed Core Thermohydraulics

The model validation program showed that the detailed calculation of thermal conditions
in each individual core cell was necessary in order to faithfully reproduce the 3D flux
distribution results of the design codes, and consequently - and even more importantly -
the global core reactivity.

The precision of the reactivity results (e.g., critical boron concentration as a function of
power level, or at low coolant temperatures) was significantly improved by the cellwise
calculation of core thermohydraulics. The same applies concerning control rod worth and
mutual shadowing.

A detailed, albeit simplified, dynamic calculation of core thermohydraulics was therefore
implemented, with the following features:

- calculation of the following variables for each individual 3D core cell:
  - coolant enthalpy.
  - coolant temperature.
  - coolant density (specific volume, void fraction).
  - fuel temperature.

- detailed calculation of the thermal conductance of the pellet/cladding gap for
each 3D core cell.
  Again, this feature proved necessary, in the Siemens case, in order to obtain the
correct fuel temperature value (and therefore the correct local fluxes, and also
global reactivity) for each individual core cell, at different power levels.
Indeed, Siemens uses a quite complex gap conductance model, which required
cellwise simulation as a function of local burn-up and power density.

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4. Model Validation Procedure

The model validation procedure was the object of detailed analysis and discussion between all the parties involved, viz.:

- the Customer, i.e., KSG and the respective Utilities.
- the Customer consultants.
- the independent Core Data suppliers.
- TT&S.

The computer codes applied for the generation of the Core Data and the validation references were fully 3D codes:

- Siemens AG and PreussenElektra AG: MEDIUM-2.
- IKE: RSYST3.

The following test categories were defined:

a) "separate-effects" tests.
b) global tests in stand-alone mode, under normal operating conditions.
c) global tests in coupled mode, under normal and off-normal operating conditions.

The respective tests are described in the following.

4.1 Stand-alone "separate-effects" tests

These tests were designed to ensure that each individual effect was correctly taken into account. Even though the tests inevitably appear artificial, and actually so they are, they were nevertheless extremely helpful in ensuring that all the calculations were individually correct, and that coincidental error masking by compensation was not taking place, for example in the rather delicate matter of temperature effects.

The following tests were performed:
- stand-alone kinetics tests.
- individual feedback effects (only one parameter varied at a time):
  - moderator temperature.
  - moderator density.
  - fuel temperature.
  - boron concentration.
- fission product poisoning.

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4.2 Stand-alone global tests

These tests were performed using the complete core model (neutronics and detailed thermohydraulics) for fixed values of the inlet core coolant temperature:

- operating conditions at several power levels: critical boron concentration, 3D flux and power distributions, coolant and fuel temperature distributions.
- critical boron concentration at zero power, for several coolant temperatures, down to 50°C.
- integral control rod worth at different temperatures and for several rod configurations.
- integral control rod worth at different power levels.
- differential control rod worth as a function of insertion depth.
- reactor scram.
- control rod drops.
- xenon axial and azimuthal oscillations.

4.3 Coupled global tests

These tests were performed during the Acceptance Test phase in the complete simulator environment, the Core model being coupled to the RCS model and the control rod drives. The specific operating procedures for the reference plant were strictly applied:

- normal start-up and shut-down procedures.
- core physics tests, according to the operating procedures.
- load changes; control rod insertion vs. reactor power.
- Incore vs. Excore evolution of axial offset.
- integral and differential control rod worth at different plant conditions.
- control rod shadowing on Incore detectors.
- loss of a reactor coolant pump and recovery without scram.
- control rod drops, faulty insertion/withdrawal, ejection.
- xenon oscillations.
- LOCA.
- ATWS.
### 4.4 Model Results

#### 4.4.1 Critical Boron Concentrations (Borssele, BOC)

<table>
<thead>
<tr>
<th>Reactor power %</th>
<th>Coolant Temperature °C</th>
<th>Critical Boron Conc. ppm</th>
<th>Critical Boron Conc. ppm</th>
</tr>
</thead>
<tbody>
<tr>
<td>0</td>
<td>50</td>
<td>1431</td>
<td>1422</td>
</tr>
<tr>
<td>0</td>
<td>150</td>
<td>1437</td>
<td>1429</td>
</tr>
<tr>
<td>0</td>
<td>250</td>
<td>1438</td>
<td>1434</td>
</tr>
<tr>
<td>0</td>
<td>292</td>
<td>1418</td>
<td>1418</td>
</tr>
<tr>
<td>3</td>
<td>292</td>
<td>1375</td>
<td>1378</td>
</tr>
<tr>
<td>50</td>
<td>304</td>
<td>1098</td>
<td>1098</td>
</tr>
<tr>
<td>100</td>
<td>304</td>
<td>980</td>
<td>980</td>
</tr>
</tbody>
</table>

#### 4.4.2 Moderator Temperature Effect (Borssele, BOC)

<table>
<thead>
<tr>
<th>Reactor power %</th>
<th>Coolant Temperature °C</th>
<th>Reactivity Effect pcm/°C</th>
<th>Reactivity Effect pcm/°C</th>
</tr>
</thead>
<tbody>
<tr>
<td>0</td>
<td>50</td>
<td>2.8</td>
<td>3.6</td>
</tr>
<tr>
<td>0</td>
<td>200</td>
<td>3.4</td>
<td>3.2</td>
</tr>
<tr>
<td>0</td>
<td>250</td>
<td>0.8</td>
<td>0.6</td>
</tr>
<tr>
<td>0</td>
<td>292</td>
<td>-5.3</td>
<td>-5.7</td>
</tr>
<tr>
<td>100</td>
<td>304</td>
<td>-16.8</td>
<td>-20.5</td>
</tr>
<tr>
<td>100</td>
<td>330</td>
<td>-40.9</td>
<td>-42.5</td>
</tr>
</tbody>
</table>
### 4.4.3 Control Rod Worth (Borssele, BOC)

#### 4.4.3.1 All rods

<table>
<thead>
<tr>
<th>Reactor power %</th>
<th>Coolant Temperature °C</th>
<th>Control Rod Worth pcm Reference</th>
<th>Control Rod Worth pcm Model</th>
</tr>
</thead>
<tbody>
<tr>
<td>0</td>
<td>50</td>
<td>-5150</td>
<td>-5130</td>
</tr>
<tr>
<td>0</td>
<td>150</td>
<td>-5760</td>
<td>-5740</td>
</tr>
<tr>
<td>0</td>
<td>250</td>
<td>-6650</td>
<td>-6670</td>
</tr>
<tr>
<td>0</td>
<td>292</td>
<td>-7180</td>
<td>-7220</td>
</tr>
<tr>
<td>50</td>
<td>304</td>
<td>-7560</td>
<td>-7410</td>
</tr>
<tr>
<td>100</td>
<td>304</td>
<td>-7480</td>
<td>-7380</td>
</tr>
</tbody>
</table>

#### 4.4.3.2 Partial rod configuration (D1 + D2 rod banks)

<table>
<thead>
<tr>
<th>Reactor power %</th>
<th>Coolant Temperature °C</th>
<th>Control Rod Worth pcm Reference</th>
<th>Control Rod Worth pcm Model</th>
</tr>
</thead>
<tbody>
<tr>
<td>3</td>
<td>292</td>
<td>-1890</td>
<td>-1880</td>
</tr>
<tr>
<td>50</td>
<td>304</td>
<td>-1950</td>
<td>-1940</td>
</tr>
<tr>
<td>100</td>
<td>304</td>
<td>-1960</td>
<td>-1930</td>
</tr>
</tbody>
</table>
4.4.4 Flux Distribution (Unterweser, EOC)

![Graph showing flux distribution in KKU - EOC Fuel assembly CO4](image)

- **Graph Details**
  - **Label**: KKU - EOC - Fuel assembly CO4
  - **Y-axis**: Normalized 3D power density
  - **Legend**:
    - Nuclear Log (real plant)
    - Model
  - **X-axis**: Aeroball measurement positions

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5. Acknowledgements

The successful implementation of the Core model for the D50 simulators was very much a collective effort, and one of its most rewarding aspects was the outstanding collaboration between all partners involved in the project.

The authors wish to express their appreciation of this collaboration to Messrs. Puschel (KKU), Lamprecht (GKN), van Bloois (KCB), Lukas (KWO), and Dr. Quast (KSG).

For many enlightening discussions, and the supply of Core Data of the best possible quality, we would also like to thank Drs. Berger, Winter, and Böhm (Siemens AG), and Drs. Bernnat and Lutz (IKE).

We also wish to acknowledge the active collaboration of Dr. W.M. Schikorr, consultant for KSG, who provided both a rigorous assessment of the model and a continuous input of much appreciated advice.

References

The Use of SIPA 2 Simulator for Safety Studies
Experience Feedbacks and Future Developments

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Abstract

SIPA 2 experience feedbacks from the beginning of its use at IPSN in 1991 and trends for the next five years are presented.

The simulator has been used for three applications:

- training of engineers working in safety analysis,
- preparation of national crisis drills,
- safety studies.

In each application, experience feedbacks are analysed to show encountered advantages and difficulties.

Trends for the next five years are:

- extension of the engineer training program (new training courses about normal operating conditions or about beyond design basis accidents),
- improvements in the validation of simulation configurations (in particular comparison with Cathare 2 new version results)
- increase of the simulation scope in connection with the SCAR project (taking into account the current power plant datapackage, the improvement of thermalhydraulic models, the extent of the system representation, new neutronic models and description of severe accident conditions).

For each trend above, a detail of the planned actions is given.

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1. Introduction

SIPA simulators were built between years 1987 and 1991. It was the result of EDF/SEPTEN and CEA/IPSN collaboration. The main builder was THOMSON. These simulators were presented at that time in references 1 and 2.

At the end of that period, two simulators were operational, one SIPA 1, set up at EDF/SEPTEN (Villeurbanne, a town close to Lyon city) and the other SIPA 2, installed at CEA/IPSN (Fontenay-aux-Roses, a town localized at ten kilometers south of Paris). A simulator unit represents about 3 million data processing lines (including 1.8 million in FORTRAN, 0.65 in ADA and 0.45 in C language). In each simulator, configurations of french 900 MW and 1300 MW PWR (respectively called CP1 and P4) can be assembled. An example of the used meshing for primary and secondary coolant systems in case of 1300 MW PWR is shown on figure 1.

After six month of tests before delivery in 92, SIPA 2 was used from 1992 to 1996 for three applications:

- training of engineers from IPSN, CEA, or other agencies,

- help in the organization of accidental crisis (definition of drill scenarios, verification of simplified computer codes...),

- realization of studies supporting safety analysis for french PWR.

One purpose of this presentation is to indicate how these three different uses were driven in the last four years, what were the encountered difficulties and the learnings for future developments.

Moreover, SIPA simulators, and SIPA 2 in particular, have to be continuously adapted to integrate the combination of new capacity in the computer tools and better models in thermalhydraulic or neutronic codes(respectively for instance CATHARE 2 or CRONOS 3D french codes). In the long term, actual planned adaptation of SIPA 2 ends this presentation, taking into account new possibilities of computer tools, code models and feedbacks of simulator uses during the last four years.

2. Training use

2.1 Training use from 92 to 96 year

The organization of a training on simulator was something relatively new for IPSN.
The needs are also different from current utilities whose purposes are to increase the ability of operator in control room to face incidental or accidental situations and in particular to apply the right procedure.

At IPSN, the trainees are engineers working:

- directly in safety authority offices,

- as experts to advise safety authorities about all technical questions happening in French PWR,

- in support of expertise, carrying out safety studies.

They are generally new in safety analysis field and have often a lack of experience in a PWR's behaviour. So, simulator training is particularly useful for them since it represents the running of a real plant. In addition, most of those engineers are working in the site of Fontenay-aux-Roses and a training on SIPA 2 is very convenient for them.

In a first step, a one week training was organized on SIPA 2. This course was devoted to the explanation of the main accidental scenarios (LOAs of Coolant Accident, called LOCA, Steam Generator Tubes Rupture, called SGTR, and Steam Line Breaks, called SLB) and to behaviour differences between 900 MW and 1300 MW PWRs. The contents of the training, called SP1, are divided in two parts:

- the first part is composed of theoretical courses made by experienced engineers working in safety analysis,

- the second part consists of SIPA 2 applications; these applications are directly prepared by the engineer team in charge of the simulator running.

A description of the program session is given in Table 1. Taking into account capacity of the simulator, training of eight engineers was possible in each session (two sub groups of four trainees).

Taking the time of about one year which was needed to prepare in advance the training contents, a one week session could be organized in 94. Years after, about three sessions were organized each year. In 97, five sessions are planned. But for this last year, another training session SP0 is proposed to the trainees (see next paragraph).

2.2 Feedbacks from 4 year training experience

Engineer trainees were generally satisfied of the SP1 training and they claim an increase of the session number and a diversification in their purpose.
Actually, a greater number of engineers are asking for that kind of training not only coming from IPSN but also from DSIN (Direction de la Sûreté des Installations Nucléaires - Nuclear Installations Safety Directorate), part of the Ministry of Industry.

Another request from the trainees was to increase the training topics to be proposed in additional sessions, as:

- plant behaviour in normal running (before studying accidental running),
- plant behaviour in severe accidental scenarios happening, for instance, beyond design basis accidents.

2.3 Planned actions for training purposes

It was decided to answer to trainee requests, but, due to the needed preparation work behind, to introduce some delay in the realization.

Concerning the diversification of sessions, a new session dealing with plant behaviour in normal running, called SP0, is now well in progress. A first session was made in march 97 and second session is planned this year. The contents consist of the initiation to standard running states (mainly those which can drive to some running difficulties) and to operational technical specifications (Specifications Techniques d'Exploitation). A description of the session program is presented on table 2.

For the second kind of session dealing with plant behaviour in beyond design basis accidents, called SP2, the aim is to organize a first session with trainees at the second semester of 98 with a preparation phase planned to start at beginning of that year.

Concerning the increase of training session number, a target of six sessions a year is aimed in 98. Taking into account the diversified training offered to engineers and the number of five sessions planned already in 97, this target seems realistic.

3. Help in the management of emergency situations

3.1 The different kinds of SIPIA 2 utilization

Three kinds of use were carried out during the last four years:

- comparison studies with simplified computer tools used by experts during a crisis,
- preparation of crisis drill accidental scenarios,

- training sessions for expert engineers who may be called in case of accidental event.

Concerning the first use, it is reminded that IPSN developed a simplified code system (called SESAME) to help diagnosis and prognosis of experts during an accidental PWR crisis. These tools must be qualified in respect to their use (often, an order of magnitude is only sufficient in that case).

The comparison with SIPA 2 results were performed for twenty transients concerning mass flow rate at pipe breaks, time delay before core uncovering in case of loss of coolant, steam generator tube break and steam pipe break scenarios. This comparison has shown the capability of the simulator to realize quickly a large amount of realistic transients (taking into account regulation and protection systems) and allowed to be better confident in SESAME code system. It revealed also, for some SGTR and SLB scenarios, the validity limits of SIPA 2 implemented models (see next paragraph and paragraph 4).

The preparation of crisis drill accidental scenarios is the greatest work of crisis drill preparation. The number of crisis drill concerning 900 MW and 1300 MW french PWR is about five drills a year. These drills involve all the national parties which will be concerned in case of real crisis in particular:

- Ministry of Internal Affairs,

- Ministry of Health,

- Nuclear Installations Safety Directorate (DSIN - part of the Ministry of Industry) and his technical support IPSN (the IPSN emergency response technical center is localized in Fontenay aux Roses site),

- EDF (french utility).

The drill scenario is prepared by a common team including EDF and IPSN engineers. The definition of the scenario or the check of its realism is performed with the help of EDF full scope and IPSN or EDF SIPA simulators. The second check option is generally adopted because present french simulators cannot simulate up to now a crisis scenario completely: the simulation stops at the beginning of core degradation.

During the last four years, SIPA 2 was used for this purpose about eight weeks a year, one drill scenario needing 2 or 3 weeks.

The third use of SIPA 2 for accidental crisis purpose is the training of IPSN engineer experts who may be called at the IPSN crisis center in case of accidental event. These experts follow periodically a teaching about crisis organization, diagnosis and prognosis methodology for assessment of radioactive transfer in the plant and the
surroundings with radiological impacts. Teaching includes a drill where experts are put in a similar situation as in a real crisis.

Accidental scenarios for the drill are on-line or recorded simulations performed with SIPA 2. Two weeks a year of SIPA 2 use were needed to perform that kind of teaching.

3.2 Feedbacks from crisis use

Those feedbacks are different for each specific use.

For the comparison of SIPA 2 results with SESAME code system, the users were particularly interested in the good description of the simulated plant, not only primary and secondary coolant system but also all around systems, regulations and protections which do not exist in batch codes. But this use revealed also the limits of SIPA 2 validity, mainly when two-phase flows happen in secondary coolant system during transients as SGTR and SLB. Actually, the homogenous flow model implemented in CATHARE-SIMU and applied for secondary system is inadequate to treat completely some of this type of transients. This issue is a more general problem connected with the qualified domain simulation scope of SIPA 2. It is treated in more details further (next paragraph 4).

The performers of crisis drill scenarios are generally satisfied from the help provided by SIPA simulator: as said already before, this tool can give a check of the realism of imagined scenarios.

Meanwhile, simulators (full scope or SIPA) have a limited field application due to the assumption of no core degradation. That limit is really a lack for this use because practically, all drill scenarios drive to core degradation in order to get a large amount of fission products released in the containment and which may be transferred outside. So, the check with simulators is only carried out for first phases of crisis scenarios.

Another boundary use appeared in the experience of teaching drill to crisis experts. The on-line simulation used in such drill is useful as it allows interactive action between learning experts and operator driving SIPA 2. But experience shows that reliability of SIPA 2 (in particular its connections with SESAME codes) is insufficient to a on-line use: sometimes, unforeseen interruptions during a drill can erase completely the expected teaching.

3.3 Planned actions for crisis use

Some actions to reduce the limitation in SIPA 2 physical models are in progress, even if they are not completely satisfactory.
Concerning the simplified two-phase flow model applied in the secondary coolant system, a drift model is now introduced in CATHARE-SIMU. After qualification tests in progress, this model will be operational next November for comparison with SESAME codes (SGTR and SLB transients) and other safety studies (see paragraph 4). Meanwhile, a completely satisfactory model (six equations two-phase flow model) will be only available in year 99 in connection with actions of the SCAR project (a description of these project actions will be done in the next paragraph 4).

Concerning the needs for checking drill crisis scenarios, simulations including core degradation are looked for. In a first step, VULCAIN code, with models describing the loss of core geometry after melting (this code is a part of ESCADRE code system developed by IPSN), was introduced as independent module not connected to CATHARE-SIMU. The connexion between the two codes is considered also but not as a first priority. In the short term, the effort will be reduced to maintain up to date an independent VULCAIN module inside SIPA 2.

In the same way, due to other actions considered as greater priority (see next paragraph 4), the on-line utilization of SIPA 2 for learnings of crisis experts will be reduced. Recorded scenarios, even without possible interactive actions, can afford good lessons for experts and appear as a safer use for that kind of application.

4 Safety studies

4.1 SIPA 2 use during the last four years

This application is the main part of the SIPA 2 use and perhaps the most original one. It takes about 20 to 25 weeks a year.

The studies can be divided in three different topics:

- the first one is concerning thermal-hydraulic accidental studies as Loss of Coolant Accident, pipe break in the secondary coolant system, loss of heat sink, ATWS; these studies needing about ten to thirteen simulator weeks are mainly connected with Probabilistic Risk Assessments (PRA) studies,

- the second one is feedback analysis of incidental events happening on French PWR; that use covered about 5 to 7 simulator weeks a year,

- the third one is connected to the verification of operating procedures; the aim of these studies is to check the validity of procedures drawed up by utilities; that use needs about 4 to 6 simulator weeks.

A study is prepared a sufficient time before its application in SIPA 2 (about one month). For that purpose, a small team of two engineer is set up, one from the running
group of SIPA 2 and the other one from the engineer department asking the study. The main job of the team is to understand correctly the study constraints (for instance data needs, definitions of boundary conditions ...), to check the capability of the simulator to carry out the study and if necessary to modify slightly the simulated reference configuration of the reactor.

It has to be emphasized at this stage, and it is too the role of the two-engineers team, that the study performed on simulator cannot replace completely, in particular for thermal-hydraulic studies, a similar study carried out with batch code as CATHARE 2. In fact, these two tools are complementary and a methodological use of each of them for a given study could be the following:

- a first simulation of the studied case in SIPA 2; this simulation allows with a minimum of work to represent all elementary systems, regulations and protections of the reactor, meanwhile, taking into account the physical models not completely up to date (in SIPA 2, the thermal-hydraulic models for primary and secondary circuits are based on CATHARE 1 batch code which has not the level of validation of present day CATHARE 2 batch code) and the limited meshes describing elementary systems, the results obtained from this first simulation must be considered as a roughing out,

- In a second step, an assessment with the present day CATHARE 2 code could be performed, assessment taking into account, in one hand, boundary conditions coming from results of the first step simulation and, in the other hand, a more detailed meshing for particular zone interesting the study,

- after the two previous steps, an iterative procedure can be carried out, depending on the accuracy claimed by the user.

It is clear that above methodology has to be adapted in function of the performed study. Actually, some studies can be limited to the first step, for instance those aiming to check operating procedures written by utilities.

4.2 Feedbacks from the SIPA 2 use for safety studies

The methodology presented in the previous paragraph was not perfectly applied during the last four years. In fact, at the beginning and even later, simulator users (those from SIPA 2 exploitation team and from engineering department asking the studies) looked at the SIPA 2 tool without a sufficient critical eye, undoubtedly impressed by the coherent pictures delivered by this big tool.

Several points of amelioration appeared to be necessary:

- a need for a better CATHARE-SIMU qualification because this module is sometimes insufficient to describe studied situations at the present day; users want in particular a better knowledge of the validity domain of models introduced in that code,
- a documentation adapted to user needs; that documentation was written by simulator maker and it is necessary to yield it more didactic for users,

- a better organization of the two engineer team; the experience shows that organization cannot be leaved to the only initiative of the two engineers but have to be formalized,

- an up-dating of the present simulator configurations to take into account the modifications introduced in reactors since year 87, actually, the simulator configurations have not be modified since that date and it is an inconvenience for studies made for PRA.

4.3 Planned actions for safety studies

To perform the needed ameliorations, a program was established with the following steps:

- a first qualification campaign is in progress; the aim of the campaign is to perform a comparison between SIPA 2 results (based on CATHARE-SIMU code, including two phase flow drift model for secondary coolant system) and up to date code CATHARE 2 for main accidental transients; a choice of five transients has been made, 3 for 900 MW and 2 for 1300 MW PWR, and this corresponding work will be finished at the end of the year; up to now, the results for the three 900 MW scenarios (LOCA, SGTR and loss of main and auxiliary feed water SG systems) are satisfactory as it can be seen on figures 2 and 3 showing the comparison results for respectively a loss of coolant coming from a two inch pipe opening and for a break of ten tubes inside one steam generator,

- the first qualification campaign will be followed up in 98 with additional accidental transients (a target of 10 transients for each reactor type is aimed); it is also planned to keep up to date these transients with the purpose to use them for later qualification campaigns after significant modifications of simulator configurations,

- the adaptation of documentation for users will start at the beginning of next year; the next fourth trimester will be devoted to the detailed definition of the user needs,

- the formalization of the job of two engineer team is underway; it is planned to test this first procedure during the last trimester of this year,

- the up-dating of the present simulator configurations is also in progress; the work is begun with 1300 MW PWR; it is an heavy job since implicating a large number of subroutines; at this day, it is planned to introduce this up-dating at the end of the first semester of 98; the work will be pursued by the up-dating 900 MW PWR with the aim to deliver this configuration to users one year later.
5 Long term developments of SIPA 2

A simulator as SIPA 2 must follow as close as possible evolutions of the basic batch codes, thermalhydraulic or neutronic, and progress of computer performances. The challenge, in that case, is to maintain actual time in the computation duration with more sophisticated code models.

On the other hand, it appears new needs from users to enlarge the simulator scope. This need was raised already in the third paragraph with the introduction of core degradation model inside simulations. But this need is also perceptible for safety studies.

Developments corresponding to the first need can be divided in two parts:

- a first part consists of the introduction of CATHARE 2 code directly inside SIPA 2; this development is included in the SCAR project (Simulator CAthare Release), which is driven in common with EDF/SEPTEN,

- a second part will consist in the implementation of a 3D neutronic code (CRONOS 3D from CEA/DRN) in order to put at the same level thermalhydraulic and neutronic models.

For these developments, the first priority was put on the first part included in the SCAR project. It is planned, after 2 years work, to set out a simulator configuration of a 900 MW PWR including CATHARE 2 models for primary and secondary coolant systems (but without the Residual Heat Removal System, called RRA, this system being still calculated with a one-phase flow model). Another step, needing three years additional work, will drive to a simulator configuration of a 900 MW PWR, including RRA system calculated with all CATHARE 2 models.

In the same way, with one year delay, a simulator configuration of a 1300 MW PWR including CATHARE 2 will be carried out.

In parallel with introduction of CATHARE 2 code, a spread of the applied field of this code will be pursued, in particular the capability to treat cold shutdown situations. These developments are also a part of SCAR project.

The above simulator configurations will be running with monoprocessor computer. Actually, it appears that a delay duration in the computational time (by comparison to the actual time) can be acceptable for IPNS needs if that delay is not too large (a limit of 2 or 3 actual time could be acceptable). Meanwhile, an introduction of some parallelization technics could be necessary (a part of this work is included in SCAR project) if this aim cannot be reached with the progress of monoprocessor performances.

Concerning the 3D neutronic development, it was decided to delay it, taking into account the disponibility in human and financial means at disposal. That development seems also less urgent due to the fact that thermalhydraulic description of the core is only
1D. In the short term, an improvement of the present 1D neutronic model in SIPA 2 will be looked for.

Developments corresponding to the second need are at the present day to the stage of statement users. They want a better description in simulator configurations for:

- heat sink elementary system,
- power supplies (normal and safety ones),
- ventilation circuit.

Preliminary work is necessary to have a better understanding of detailed needs. Meanwhile, taking into account the amount of actions needed for developments already decided, and described in paragraphs 3 and 4, it is clear that a significant work on those subjects cannot be performed before two years.

6. Conclusion

Feedbacks of the SIPA2 use were presented and lessons made for future developments.

An important program of actions is in progress to improve the use of the simulator mainly for safety studies.

On the short term, this program consist of:

- the qualification of actual simulator configurations by comparison with CATHARE 2 code batch results,
- the formalization of the work of the two engineer team, work for preparation, following and finalization of a safety study on simulator,
- the adaptation of the user documentation,
- the up to date simulator configurations taking into account modifications on french PWR since 87.

On the long term, the main development concerns the introduction of CATHARE 2 inside simulator configurations. This development is partly included in the SCAR project.
References

1) SIPA 1 (EDF) and SIPA 2 (IPSN): two Post-Accident Simulators for training and engineering; L. Bregéon, J. M. Delbecq, A. Géranton, M. Rangdet, H. Sureau, Revue Générale Nucléaire n° 4, July and August 1992.


Table 1: SP1 training contents

<table>
<thead>
<tr>
<th>Week day</th>
<th>Theoretical course contents</th>
<th>Practical application on SIPA 2</th>
</tr>
</thead>
<tbody>
<tr>
<td>Monday</td>
<td>general presentation, main physical phenomena</td>
<td>general application on simulator</td>
</tr>
<tr>
<td>Thursday</td>
<td>LOCA studies</td>
<td>practical application n°1</td>
</tr>
<tr>
<td>Wednesday</td>
<td>900 MW/1300 MW differences</td>
<td>practical application n°2</td>
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<tr>
<td>Tuesday</td>
<td>SGTR studies</td>
<td>practical application n°3</td>
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<td>Friday</td>
<td>SLB studies</td>
<td>practical application n°4</td>
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Table 2: SP0 training contents

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<th>Morning</th>
<th>Afternoon</th>
</tr>
</thead>
<tbody>
<tr>
<td>Monday</td>
<td>Theoretical course contents 1: auxiliary systems, permissive functions, plant standard status</td>
<td></td>
</tr>
<tr>
<td>Tuesday</td>
<td>Theoretical course contents 2: main regulations and shutdown operation</td>
<td>Theoretical course contents 3: criticality and power operation, 900-1300 MW differences</td>
</tr>
<tr>
<td>Wednesday</td>
<td>Practical application 1: heatup from cold shutdown to hot shutdown</td>
<td>Practical application 2: criticality and power operation</td>
</tr>
<tr>
<td>Thursday</td>
<td>Practical application 3: power transients and cool down</td>
<td></td>
</tr>
</tbody>
</table>
Figure 1. Mesh used in SIPA 2 to described Primary and Secondary coolant system of a 1300 MW PWR.
Figure 2

Loss of coolant for two inch opening - Pressure evolution of the Primary system

Figure 3

Break of ten tubes inside Steam Generator - Pressure evolution of the Primary system

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A VERIFICATION AND VALIDATION PROGRAM
FOR SIMULATORS FOR
SOVIET-DESIGNED NUCLEAR REACTORS

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ABSTRACT

Brookhaven National Laboratory is providing technical support to the U.S. Department of Energy as part of a program to develop and implement training simulators at nuclear power plant (NPP) sites in Russia and Ukraine. This program includes both full-scope and analytical simulators for VVER-440 and VVER-1000 reactors; these simulators will be incorporated into the staff training programs at each site. As part of the procurement and delivery of each simulator, a verification and validation (V&V) program has been developed for each of the simulators. This program includes: a training program, presented to the staff at each NPP site, on the verification and validation tasks; completion of the verification and validation procedures during the simulator acceptance test; and the final acceptance of the simulator at the NPP site. These V&V activities are discussed in this paper.

1. INTRODUCTION

The U.S. Department of Energy is implementing a program to provide training simulators to nuclear power plant sites in Russia and Ukraine. These simulators will be incorporated into the development of operational staff training programs at each site. There are currently eleven active simulator programs (both full-scope and analytical) in Russia and Ukraine. As noted in the Table, some projects are being completed jointly with the active

1This work was performed under the auspices of the U.S. Department of Energy, Contract No. DE-AC02-76CH00016.

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participation of the NPP site. This paper provides an overview of a verification and validation program that has been developed for implementation during each of the simulator projects.

2. DESCRIPTION OF THE PROGRAM

In the United States and other Western countries, it is well recognized that a detailed procedure for V&V tasks should be followed during the acquisition of a training simulator for a nuclear power plant (NPP). In addition, as part of maintaining the simulator fidelity and for the simulator to remain a useful training facility, a formal procedure for the maintenance and upgrading of the simulator should be followed. Western specialists, recognizing the importance of these procedures, provide extensive training in the procedure implementation. Thus, the incorporation of similar training programs into the simulator projects for Russia and Ukraine was considered important. Since the NPP staff are key participants in the V&V activities for each simulator project, training courses were developed and offered to the NPP staff. Simultaneously, V&V activities were incorporated into the Acceptance Test Procedures (ATP's) for each of the simulator projects. For the staff of the Khmelnitovsky NPP, these training activities have been completed, the ATP has been developed and approved, and the Acceptance Test is currently being performed.

Based on detailed input from the Brookhaven National Laboratory (BNL) staff, the training course on the “Verification and Validation of Nuclear Power Plant Simulators” was developed by the General Physics Corporation. In view of the large differences in the “training culture” between the NPP staff from the formerly Soviet countries and the Western countries, it was considered important to obtain the feedback of some Ukrainian NPP staff to this training material before formally presenting this course to NPP staff. Therefore, a one-week workshop on Verification and Validation procedures for nuclear power plant simulators was held at the Brookhaven National Laboratory during April 1997. One staff member each, from the Khmelnitovsky and the South Ukraine NPPs and from the Engineering Technical Center in Kiev were invited to the workshop. Based upon the feedback received from the participants, the course material was modified. An outline of the course material is given in the next section. The course material was translated into Russian, and it was recently presented at the Khmelnitskyy NPP in Ukraine. In addition to the Khmelnitskyy NPP staff, staff members from other Ukrainian NPP’s participated in this training course. The course was well received by the Ukrainian participants.
3. SUMMARY OF THE V&V COURSE PRESENTED TO NPP STAFF

The training course is divided into the following main sections:

(1) Experience of U.S. utilities in the verification and validation of full-scope simulators.

The topics covered in this section include: the purpose of a training simulator, definitions of verification and validation, cooperation with other simulator operators, Procedures (e.g., INPO Good Practice guideline TQ-504), the Simulator Acceptance Test Team, and the Simulator Maintenance Test Team.

(2) Simulator Verification.

The topics covered in this section include: Reference plant data (design data, performance data), alternate data sources, assumptions and simplifications, gaps in data, design reviews physical configuration (hardware, facilities, systems), importance of human factors, limits of simulation, instructor station features, and importance of accuracy.

(3) Documentation for Verification and Validation

The topics covered in this section include: design data base, Acceptance Test Procedures, the simulator specification, malfunction cause and effects, control room fidelity report, operating procedures, and the configuration management system.

(4) Simulator Validation

The topics covered in this section include: individual system tests and integrated hardware and software tests, vendor test report (completed copy of the ATP, summary sheet and copies of the Discrepancy Reports, summary list of all assumptions and simplifications), Factory Acceptance Test, simulator transients scenarios, malfunctions, simulator capabilities, and on site acceptance testing.

(5) Potential events and systems to be considered during FAT and SAT.

This is the discussion of scenarios that are deemed to be important based upon training requirements and Probabilistic Risk Assessment (PRA) studies.

(6) Validation and Verification Procedure

The topics covered in this section include: simulator certification report procedure and a detailed step-by-step example of a verification and validation procedure followed at a U.S. NPP.
4. Summary of ATP Verification and Validation Activities

The verification and validation activities are performed during the Acceptance Test Procedure. The following outline of selected V&V tasks has been obtained from the ATP prepared for the Khmelnitskyy NPP simulator. The acceptance test at Khmelnitskyy is currently being performed.

1. Control Room Hardware and Configuration Verification

   This activity confirms that the specification requirements for the control room hardware are satisfied. The hardware verification includes the control panels and consoles, the remote shutdown panels, the input/output systems, the display systems, and all wiring and grounding.

2. Computer Hardware (including the simulation computer) Diagnostic Verification

   This task verifies the operability of the computer and simulator vendors’ diagnostic test software. Also included is the determination of the spare time and memory of the simulation computer.

3. Instructors Station Testing

   This activity provides an extensive test of the functions required from the Instructor Station. Included are tests of the simulator control functions, the training control functions, and the training documentation functions.

4. Training Mission Tasks

   Verified during this step are reactor operational evolutions that the simulator is required to represent. Included are reactor startup, shutdown, and power rundown.

5. Simulator Malfunctions

   This task verifies each of the malfunctions that are identified in the simulator specification. Component malfunctions are included.

6. Transient Execution Tests

   The activity provides an extensive test of the reactor transients that are specified in the specification. The results are reviewed by the NPP operational staff and other subject matter experts participating in the test.

The V&V tasks listed above have been selected from the Khmelnitskyy NPP Acceptance Test procedure. The complete procedure provides a complete verification of the simulator specification requirements.
5. SUMMARY

In summary, a comprehensive verification and validation program has been developed in support of the training simulators that are being procured for Russian and Ukrainian NPP sites. Since the execution of the V&V procedures involves the participation of Russian and Ukrainian specialists, an extensive training program on the preparation and execution of the procedures has been developed and presented. The first application of this program is currently in progress at the Khmelnitksyy NPP where the simulator acceptance test is being performed.

Table: Summary of Simulator Projects in Russia and Ukraine

<table>
<thead>
<tr>
<th>NPP</th>
<th>Simulator Type</th>
<th>Procurement Responsibility</th>
<th>Project Initiation</th>
</tr>
</thead>
<tbody>
<tr>
<td>Balakovo VVER-1000</td>
<td>Analytical</td>
<td>U.S.</td>
<td>May 95</td>
</tr>
<tr>
<td>Bilibino</td>
<td>Analytical</td>
<td>U.S. and NPP</td>
<td>Jan. 97</td>
</tr>
<tr>
<td>Chernobyl Unit 3 RBMK</td>
<td>Analytical</td>
<td>U.S.</td>
<td>Dec. 95</td>
</tr>
<tr>
<td>Kalinin Unit 2 VVER-1000</td>
<td>Full-Scope</td>
<td>U.S. and NPP</td>
<td>May 95</td>
</tr>
<tr>
<td>Khmelnitksyy VVER-1000</td>
<td>Full-Scope</td>
<td>U.S.</td>
<td>Nov. 94</td>
</tr>
<tr>
<td>Kola Unit 4 VVER-440</td>
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<td>U.S. and NPP</td>
<td>May 95</td>
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<tr>
<td>Novovoronezh VVER-440</td>
<td>Analytical</td>
<td>U.S.</td>
<td>May 95</td>
</tr>
<tr>
<td>Rivne Unit 3 VVER-440</td>
<td>Full-Scope</td>
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<td>Feb. 96</td>
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<tr>
<td>South Ukraine Unit 1 VVER-1000</td>
<td>Full-Scope</td>
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<td>South Ukraine Unit 3 VVER-1000</td>
<td>Full-Scope</td>
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<td>Zaporizhzhya Unit 5</td>
<td>Full-Scope Upgrade</td>
<td>U.S. and NPP</td>
<td>Jan 97</td>
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INTERACTIVE GRAPHICAL ANALYZER
BASED ON RELAP5/Mod 3.2 - NPA

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ABSTRACT

The work presented in this paper consists on the development of a Graphical Interactive Analyzer for Ascó (two units) and Vandellós (one unit) Nuclear Power Plants, all of them are three loop Westinghouse PWR with rated electrical power around 1000 Mwe. Basic steps are:

- Development of the thermal-hydraulic and kinetic model for RELAP5/mod3.2 corresponding to NSSS, Steam Flow paths from Steam Generators to Turbine and Condenser, Feedwater System, Emergency Core Cooling System; and related protection and control systems.

- Development of Graphical representation, for NPA-1.3.4., to permit the user interact with the model.

- Validation against experimental data.

The result is an engineering tool that can help on Plant transient analysis, and on the study of modifications proposed on the components simulated; it's also a powerful tool for operator teaching.

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1. PLANTS DESCRIPTION

1.1. ASCÓ

Ascó Nuclear Station is a nuclear power plant with two 968-Mwe Three Loop Westinghouse PWR, located in Tarragona (Spain), by Ebro river. The first criticalities were reached in June 1983 (Unit I) and September 1985 (Unit II). All major components are standard Westinghouse, except Steam Generators, that have recently been substituted by Framatome-KWU Generators.

1.2. VANDELLÓS

Vandellós II NPP, owned by Endesa (72%) and Iberdrola (28%), is located in Tarragona (Spain), by the Mediterranean Sea. It’s a Three Loop Westinghouse PWR with model F Steam Generators, and a nominal power of 2775 Mwt and 1004 Mwe. Its commercial operation started on March 1988.

2. MODEL DESCRIPTION

The scope of the model is similar in both cases, so only one model is described, making notice only on important differences.

Primary circuit. The model includes: the reactor vessel, including bypass flow paths, three separate loops with the reactor coolant pumps and primary side of Steam Generators, the pressurizer, surge line, spray lines and valves, the make-up and let-down of Chemical and Volumetric Control System, (Vandellós also has a model to control boron concentration), and the Pressurizer Relief and Safety Valves

Secondary Circuit. Including Steam Generators, Main Steam lines to turbine, steam-dump lines and valves, Relief and Security Valves for each SG.

Feedwater System. The boundary conditions are the water properties in the suction of main feedwater pumps and auxiliary feedwater pumps. So, in main feedwater system the feedwater pumps, the high pressure preheaters and their bypass lines and the flow control valves are simulated; auxiliary feedwater model includes pumps, control valves and connecting lines.

Accumulators and Safety injection.
Protection and control system. All of the signals of protection system related with the model are included, and it's possible to introduce a great quantity of external signals. The following control systems have been simulated:

Rod control and reactor kinetics.

Pressurizer pressure and level control.

Main and auxiliary feedwater control.

Turbine and steam-dump control.

Steam Generators relief valves.

Permissive and interlock circuits.

Manual actuation on control and protection systems. Almost 100 interactive variables for each model are accessible from NPA interface.

3. MODIFICATIONS TO NPA AND RELAP5

Several improvements have been incorporated to the code, in order to facilitate user interaction with the model:

3.1. Interactive Restart

A new button, labeled "RESTART", appears on command screen. This button permits the user, during a simulation, recording current state (restart record) in whatever instant of the simulation, or going back to any previously recorded state. So it is possible, from any point, to test different operations and select the better one.

3.2. Push Button Command

In order to simulate the actual situation in Control Room, where some push-buttons can eventually go back to a default position, a new function to pass commands from NPA to RELAP has been developed, in which the same action on NPA can set different values for a variable.
3.3. New ON/OFF dynamic function

This is a "non-consuming color" function that permits to increase the number of simultaneous dynamic functions, so all the masks, described in the next section, can be "active-masks"; that is, masks which are actualized with current calculations.

3.4. New thermocolor subroutine

The original subroutine has a very wide range of colors, leading in same cases to non intuitive colors. A new subroutine has been programmed limiting the range of colors:

- Liquid: Blue, darker as the liquid becomes more subcooled.
- Two-phase: Cyan, lighter as the void fraction rises.
- Vapor: Yellow, darker as the vapor becomes more superheated.

3.5. Screen Plot Improvements

3.5.1. ALIAS

This new button in PLOT menu makes a correspondence between RELAP variables and user selected names, also allows changing units and axis labels

3.5.2. COMPARE

Results can be compared with archived data (on an ASCII file), selecting variables from a new Plot menu.

3.5.3. Minor improvements

Some other changes have been incorporated on screen plots, such as:

plot title

dot plotting of external data (when selected)

new REDRAW button, that permits to incorporate new data generated by the code to an existing plot.
4. GRAPHICAL INTERFACE

The G.I. for NPA in its current state consists of four masks for Ascó model and five masks for Vandellós model, these masks permit an overview of the results of the calculation, and give the user the ability of interacting with the model in an easy way. For instance, if the user wants to open a valve, he simply picks-up with the mouse on that valve and selects the appropriate action on the menu (in this case CLOSE). The masks are the following:

Main view. It's a global overview of NSSS, where the nodes in RELAP model are animated according to their thermal-hydraulic properties. Most interactive commands are associated with this mask.

Figure 1. Main Mask for Vandellós Model
Figure 2. Main Mask for Ascó Model

- Trend Graphs of the most important variables.
- State Panel. Representing with colored indicators (red-yellow-green), the state of control and protection system.
- Feedwater System. Containing the representation of feedwater system and related interactive menus.
- ECCS. Currently only available for Vandellós model. Shows the representation of the system and also include its associate commands.
5. VALIDATION PROCESS

Both models are now undergoing a validation process. Some transients occurred in the Plants have been selected trying to involve as many subsistems in the models as possible. The most significant results are shown in the following.
5.1. ASCÓ

5.1.1. Reactor Trip at ASCÓ I

This transient was initiated by reactor trip because of low primary flow signal in one loop while the plant was working at full power. The figures show how the model reproduces the evolution of Average Temperature, controlled by steam-dump system in automatic mode and manual actuation on auxiliary feedwater system, and the evolution of primary pressure controlled with pressurizer heaters and CVCS.
5.1.2. Loss of Feedwater in a SG at ASCÓ I

The initiating event was the closure of one Feedwater Control Valve, leading to reactor trip because of low level in the corresponding Steam Generator. So this is a non symmetric transient. The trip occurs at the same time in the model, and after the trip the evolution of the plant is well simulated.
5.1.3. Load-Reject 100% - 50% at ASCO II

This is a start-up test conducted after the replacement of Steam Generators. Rod control, reactor kinetics and steam-dump models in automatic mode are the systems validated with this transient, as can be seen in the above figures.
This is a pure turbine trip that caused reactor trip without any additional complication.
5.2.2. Main Steam Isolation

In this transient, reactor trip is due to low level in one SG, originated by void collapsing after isolation. The evolution of the transient has been reproduced before and after the trip. As shown in the figure above, the effectiveness of pressurizer relief valves has also been validated.
This transient initiates with a reduction in primary flow due to electrical disturbances, the setpoint of high neutron flux rate is reached, and after a few seconds RCP’s trip. The Plant operates in natural circulation. At second 525 Main Steam Isolation Valves are closed by operator. At time=800. s one of the RCP is switched on. The results of RELAP model are in very good agreement with Plant data.
6. CONCLUSIONS

RELAP5/mod3.2 has shown as a very good code to simulate transients in ASCÓ and Vandellós NPP.

The models developed for both plants reproduce quite well the Plant behavior.

NPA is a very friendly tool both for engineering calculations and specially for interactive educational purposes.

The improvements incorporated by PMSA to RELAP5-NPA make it more powerful and are useful in validation processes and teaching sessions.
Proceedings of the 2nd CSNI Specialist Meeting on Simulators and Plant Analysers

Abstract

The safe utilisation of nuclear power plants requires the availability of different computerised tools for analysing the plant behaviour and training the plant personnel. These can be grouped into three categories: accident analysis codes, plant analysers and training simulators.

The safety analysis of nuclear power plants has traditionally been limited to the worst accident cases expected for the specific plant design. Many accident analysis codes have been developed for different plant types. The scope of the analyses has continuously expanded.

The plant analysers are now emerging tools intended for extensive analysis of the plant behaviour using a best estimate model for the whole plant including the reactor and full thermodynamic process, both combined with automation and electrical systems. The comprehensive model is also supported by good visualisation tools.

Training simulators with real time plant model are tools for training the plant operators to run the plant. Modern training simulators have also features supporting visualisation of the important phenomena occurring in the plant during transients.

The 2nd CSNI Specialist Meeting on Simulators and Plant Analysers in Espoo attracted some 90 participants from 17 countries. A total of 49 invited papers were presented in the meeting in addition to 7 simulator system demonstrations. Ample time was reserved for the presentations and informal discussions during the four meeting days.

Keywords
nuclear power plants, safety analysis, reliability, accidents, accident prevention, simulation, simulators, validation, modelling, training
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