NUCLEAR ENERGY AGENCY
COMMITTEE ON THE SAFETY OF NUCLEAR INSTALLATIONS

BEST ESTIMATE METHODS IN THERMAL HYDRAULIC SAFETY ANALYSIS

Seminar Proceedings

Ankara, Turkey
29 June - 1 July 1998
ORGANISATION FOR ECONOMIC CO-OPERATION AND DEVELOPMENT

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The mission of the NEA is:

− to assist its Member countries in maintaining and further developing, through international co-operation, the scientific, technological and legal bases required for a safe, environmentally friendly and economical use of nuclear energy for peaceful purposes, as well as

− to provide authoritative assessments and to forge common understandings on key issues, as input to government decisions on nuclear energy policy and to broader OECD policy analyses in areas such as energy and sustainable development.

Specific areas of competence of the NEA include safety and regulation of nuclear activities, radioactive waste management, radiological protection, nuclear science, economic and technical analyses of the nuclear fuel cycle, nuclear law and liability, and public information. The NEA Data Bank provides nuclear data and computer program services for participating countries.

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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 meeting.

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.
OECD/CSNI Seminar on
"Best Estimate Methods in
Thermal Hydraulic Safety Analysis"
Ankara (Turkey), June 29 – July 1, 1998

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A Contents
B Summary and Conclusions
C Papers
D Participants
# A. TABLE OF CONTENTS

<table>
<thead>
<tr>
<th>PAGE</th>
<th>B. SUMMARY AND CONCLUSIONS ....................................................................................................</th>
<th>11</th>
</tr>
</thead>
<tbody>
<tr>
<td></td>
<td>C. PAPERS................................................................................................................................</td>
<td>25</td>
</tr>
<tr>
<td></td>
<td>SESSION I: KEYNOTE PRESENTATIONS.............................................................................................</td>
<td>25</td>
</tr>
<tr>
<td></td>
<td>Chair: M. Réocreux - O. Kadiroglu</td>
<td></td>
</tr>
<tr>
<td></td>
<td><strong>NRC Regulatory Philosophy for Commercial Nuclear Power Plants</strong> ..............................................</td>
<td>27</td>
</tr>
<tr>
<td></td>
<td>A. Thadani</td>
<td></td>
</tr>
<tr>
<td></td>
<td><strong>The Developing Roles of “Best-Estimate” Thermal-Hydraulic Calculations and Uncertainty Analyses in Licensing in Canada</strong></td>
<td>49</td>
</tr>
<tr>
<td></td>
<td>D. Newland</td>
<td></td>
</tr>
<tr>
<td></td>
<td><strong>Status of the French Approaches for Using Best-Estimate Codes in Licensing</strong> ..................................................................................................................</td>
<td>69</td>
</tr>
<tr>
<td></td>
<td>M. Réocreux, P. Jamet</td>
<td></td>
</tr>
<tr>
<td></td>
<td><strong>Best-Estimate Practices in Licensing in Germany</strong> .............................................................................................................................................................................</td>
<td>81</td>
</tr>
<tr>
<td></td>
<td>R. Kirmse</td>
<td></td>
</tr>
<tr>
<td></td>
<td>SESSION II: OVERVIEW OF PAST AND CURRENT CSNI THERMAL-HYDRAULIC ACTIVITIES..........................................................................................................................</td>
<td>127</td>
</tr>
<tr>
<td></td>
<td>Chairmen: O. Yesin, F. Eltawila</td>
<td></td>
</tr>
<tr>
<td></td>
<td><strong>Activities of Principal Working Group 2 on Coolant System Behaviour</strong> .................................................................................................................................</td>
<td>129</td>
</tr>
<tr>
<td></td>
<td>M. Réocreux</td>
<td></td>
</tr>
<tr>
<td></td>
<td><strong>Best-Estimate Codes and Use of CSNI International Standard Problems</strong> .................................................................................................................................</td>
<td>143</td>
</tr>
<tr>
<td></td>
<td>N. Aksan, M. Réocreux</td>
<td></td>
</tr>
<tr>
<td></td>
<td><strong>Overview of the CSNI Separate Effect Test and Integral Test Facility Matrices for Validation of Best-Estimate Thermal-Hydraulic Computer Codes</strong> ..................................................................................................................</td>
<td>163</td>
</tr>
<tr>
<td></td>
<td>N. Aksan, H. Glaeser</td>
<td></td>
</tr>
<tr>
<td></td>
<td><strong>Thermal-Hydraulic Accident Analysis</strong> .............................................................................................................................................................................................</td>
<td>187</td>
</tr>
<tr>
<td></td>
<td>J. Misak</td>
<td></td>
</tr>
</tbody>
</table>
SESSION III - PART 1: BEST-ESTIMATE METHODOLOGIES AND ASSOCIATED UNCERTAINTIES

Chairmen: N. Aksan, H. Glaeser

Best-Estimate Methods in PWR Safety Analysis
N. A. Butt, B.J. Holmes, J.P. Rippon, M. G. Woodhill

Use of Best-Estimate Methods in a Licensing Case of 1300 Mwe PWR
A. Amri

Application of Best-Estimate Methods to LOCA in a PWR
F. Depisch, G. Seeberger, S. Blank

Application of Best-Estimate Methods for VVER Reactors in Hungary
I. Toth, C.S. Györi, T. Trosztel

Best-Estimate Methods in CANDU Reactor LOCA Analysis
D. Richards, V.S. Krishnan, J.C. Luxat

Utility Perspective on the Use of Best-Estimate Codes in the Licensing Environment
P. Garcia Sedano

The Current Status of the Safety and the Best-Estimate Analysis for BWR Stability, Transient and Accident Events Using Thermal-Hydraulic Codes
T. Fujii, T. Nakajima, T. Anegawa

SESSION III - PART II

Chairmen: V.S. Krishnan, I. Toth

Application of New Best-Estimate Code to PWR and APWR LBLOCA Analysis
S. Urata, K. Okabe

Application of Best-Estimate Methods for Optimization of EPR’s Emergency Core Cooling Mode
F. Curca-Tivig

Requirements for Best-Estimate Containment Safety Analysis
J. Rohde, B. Schwinges

Overview of Uncertainty Issues and Methodologies
F. D’Auria, E. Chojnacki, H. Glaeser, C. Lage, T. Wickett

Application of Uncertainty Methods in the OECD/CSNI Uncertainty Methods Study
H. Glaeser, T. Wickett, E. Chojnacki, F. D’Auria, C. Lage Perez

Safety Analysis: Treatment of Uncertainties
R. B. Duffey, A. Abdul-Razzak, H. Sills, B. McDonald, V.S. Krishnan, T. Andres
SESSION IV: SELECTED ISSUES OF THERMAL-HYDRAULIC SAFETY ANALYSES.............. 505
Chairmen: F. D’Auria, T. Speis

     A Case Study on Small Break LOCA............................................................................................... 507
     A. Tanrikut

     Performance Evaluation of the Emergency core Cooling System for an Evolutionary ....................... 535
     Pressurised Water Reactor 
     H. R. Choi, C.J. Choi, J.H. Choi, S.K. Lee

     RELAP5/PARCS Generalized Thermal-Hydraulics/Neutronic Interface ............................................ 563
     T. Downar, D. Barber, V. Mousseau, D. Ebert

     Three Dimensional Kinetics Coupling to Thermal Hydraulics .............................................................. 571
     Joint Effort of CEA/SERMA and CEA/SMTH 
     A. Bengaouer, G. Geffraye

     The Need of Coupled 3D Neutronics in DBA and DBA Analyses Using Conservative ....................... 581
     or Best-Estimate Approach 
     R. Kyrki-Rajamaki

     Thermal-Hydraulic Challenges in Advanced Reactor Designs ............................................................. 589
     J. M. Kelly

     Beyond Design-Based Accident (BDBA) Phenomena and Risk Reduction Issues ............................. 605
     T. P. Speis

     A New Safety Approach for Future PWRs............................................................................................ 629
     R. Kirmse, M. Champ

D. LIST OF PARTICIPANTS......................................................................................................................... 643
B. SUMMARY AND CONCLUSIONS

1. Introduction

1.1 Sponsorship
The CSNI Seminar on Best Estimate Methods in Thermal Hydraulic Safety Analysis, held on 29 June – 1 July 1998 in Ankara, Turkey, was sponsored by the Committee on the Safety of Nuclear Installations (CSNI) of the OECD Nuclear Energy Agency (NEA). It was organized in collaboration with Turkish Atomic Energy Authority (TAEK).

1.2 Background and Objectives
In the spring of 1992 an OECD Specialist Meeting on Transient Two-Phase Flow ("Current issue in system thermal-hydraulics", April 6-8, 1992) was held in Aix-en-Provence, France. Next, there was an OECD Workshop on Transient Thermal-Hydraulic and Neutronic Codes Requirements (Annapolis, USA, November 5-8, 1996). The issues raised during the meetings indicated strongly a need for well established best estimate methodology for the use in plant safety analysis. A number of other CSNI activities addressed status of codes validation and related issues of codes uncertainties.

Discussions within the CSNI have led to the conclusion that the time is ripe to organize an international seminar to review the insights from and the status of utilization of the Best Estimate methods in plant safety analysis.

The scope of this Seminar is limited to safety analysis needed in support of licensing process. Therefore, the workshop did not specifically address issues related to code development and physical models.

The objectives of the meeting were:

− to exchange information on the Member countries' methodologies used and/or required in the licensing process, and

− to discuss the licensing issues associated with uncertainties and evaluation of T/H safety margins in conservative and BE approaches.

− to provide information to Turkish hosts on the use of Best Estimate methods in support of licensing.
The Seminar was structured into the following sessions:

1. Keynote presentations
2. Overview of past and current CSNI Thermal-Hydraulics activities
3. Best Estimate Methodologies and associated uncertainties (2 Sessions)
4. Selected Issues of Thermal-Hydraulics Safety Analyses
5. Final Discussion

A Program Committee was nominated by the CSNI to evaluate the abstracts of proposed papers, to select the papers for presentation, to organize the sessions and to develop the final program of the workshop, appoint Session Chairman, etc. Its members were:

- Mr. Nusret AKSAN, PSI, Switzerland, (Chairman)
- Mr. Yilmaz BEKTUR, TAEA, Turkey
- Mr. N.A.J. (Nick) Butt, Nuclear Electric, UK
- Dr. Farouk ELTAWILA, NRC, USA
- Dr. V.S. (Krish) KRISHNAN, AECL, Canada
- Dr. Michel REOCREUX, IPSN, France
- Mr. Victor TESCHENDORFF, GRS, Germany
- Mr. Andre DROZD, NEA (secretary)

The meeting was attended by 68 participants from 16 OECD countries, OECD/NEA and IAEA. The list of participants is given at the end of the proceedings.

It should be noted that, at the time this Seminar was organized, it was not intended to publish the presentations in a proceeding. After receiving the presentations and papers during the Seminar, Program Committee (PC) agreed that the presented papers are high quality and valuable for research and licensing organizations. Consequently, PC decided to publish the proceedings of the Seminar.
2. General remarks

The safety limits of the existing plants are currently not reached. Even more, some "better" analyses reveal existence of additional safety margins. There are many possible approaches to best-estimate analysis. It is known that the current conservative approach to LBLOCA is "overly conservative" but that may not be so conservative for the other break sizes, caution need to be exercised in such cases. As the keynote speakers indicated, the use of BE methods is a "fact of life" regardless of what the regulators may decide about it. Currently only Siemens is using uncertainty analysis in the licensing process. It has been noted, that regulators are usually "evaluating" safety analysis rather than performing full scope analysis. The uncertainty methodologies exist, but need improvement in the direction of simplified usability.

The studies of beyond design basis accident (BDBA) and severe accident (SA) scenarios are very important to evaluate a level of "enhanced" plant safety. As it is difficult to calculate exactly the BDBA and SA scenarios, the "trends" of accident progressions need to be predicted as realistic as possible. Also there is a need for better coupling of thermal-hydraulic computer codes with neutronic, mechanical, containment and PRA analyses. In general, best-estimate codes are being used in a "conservative mode".

Some specific points made during the sessions:

- The safety limits are not going to be changed. Rather, the use of BE codes will change in a direction of establishing the limits and "measure" the margins,
- In the U.S.A., the "old" criteria are still used by the applicants, but BE methods are allowed supported by a PRA evaluation,
- A "common approach" of France and Germany differs in details, e.g., in France one code is used by both regulator and utilities, while in Germany different codes are used to have a better chance to uncover common cause failure,
- Public scrutiny of regulators forces to keep conservative criteria, but BE evaluation may be used,
- Assessment of codes against validation matrices helps to estimate uncertainties,
- So far, the performed comparisons between CFD and 1-D codes lead to no conclusions,
- With strong emphasis on physics, the use of computer codes may give some valuable input and support to decision making and licensing, in addition to the use of the available research results in a distilled form,
- Nodalization is equally important as modeling. A course nodalization may give a correct average, but leads to incorrect estimate of local conditions (classic examples: void and two-phase flow regimes).

Points made regarding uncertainties studies:

- German paper on uncertainties shows that fuel related uncertainties are the most important, and that confirms the original CSAU studies done by US.NRC,
- If the established safety limits are "truly conservative", a plant should be permitted to operate as close to limits as possible,
- There are many sources of uncertainties. The most basic are the use of approximate codes solved by approximate methods, and the use of questionable correlations,
- Currently there is a good understanding of similarities and differences between the uncertainty analysis methods as provided by the UMS study,
- There is no "single" parameter that describes plant safety but there are many,
Elements of Canadian approach to uncertainty analysis consists of PUA-RUA-CUA, i.e. Plant (how well we know), Representation (how well we model), and Codes (how well we "do" code analysis).

When codes are coupled (e.g., thermal-hydraulic to neutronic) there is a "transfer of uncertainties". These type of uncertainties should be addressed and they can be addressed. There are already some plans to investigate a demonstration case with the available methods, presently.

**Additional recommendations for uncertainty analysis in DBA and BDBA:**

Improve knowledge on sensitive input uncertainties;

Work on algorithm to obtain uncertain input parameter distributions from comparison of calculation results and experimental data (some work performed in USA, work is going on in France and Germany);

Check conservatism of input parameter uncertainty ranges and distributions; choose only experiments as a basis for uncertainty specification which are appropriate for the plant scenario under investigation;

Minimize need for expert judgment as far as practicable;

Uncertainty data base should be set up to store information on parameter uncertainties; although this information is partly code and application dependent, valuable information can be obtained;

A new generation of codes could provide an „internal assessment of uncertainty“;

Improve the sensitive models in thermal hydraulic codes in order to reduce code uncertainties (e.g. critical flow, interfacial shear in the core region, counter-current flow).

**Some answers to the general terms of the Seminar:**

BDBA evaluation is necessary in the German licensing process since 1 January 1994 to ensure that even extremely unlikely events involving core melt-down would not require radical actions to ensure protection against the damaging effects of ionising radiation outside the fence of the installation site.

No limit of safety margins is specified, a licensing limit is sufficient. It has to be assured that a given plant will not exceed the licensing limits. The purpose of using a best estimate code and performing an uncertainty evaluation is to provide assurance that the licensing limits will not be exceeded with a probability of 95% or more.

BDBA analysis should not be conservative when the analysis is needed for risk reduction.
The existing uncertainty methods are useful. Guidelines on the choice of methods has been given in the frame of the Uncertainty Methods Study (UMS). There is no completely general theory available. The differences in the results of various uncertainty methods are due to differences in application, mainly due to the more or less conservative specification of ranges or distributions for the uncertain input parameters. In the Uncertainty Methods Study the uncertainties were specified to the conservative side by most participants. The situation would be similar for conservative bounding analyses using best estimate codes when for some dominant parameters conservative values have to be specified.

Further remarks and conclusions are included in the summaries of the specific sessions in next following pages.
3. 

Session specific summaries and conclusions

SESSION 1:  
Keynote presentations  
Chairmen: Reocreux / Kadiroglu

Session 1 was devoted to four keynote presentations. The objectives of these presentations were to provide insights on practices in some selected countries which will give a representative description about the status in OECD countries of the use of Best Estimate thermal hydraulic codes in safety analyses and in the subsequent licensing process.

The first paper which was presented by USNRC (A. Thadani), gave a broad and historical overview of the evolution of regulatory requirements for the US commercial nuclear power plants.

As starting point of this overview, the safety studies initiated by USNRC (AEC before 1974) were first reviewed. The long story of these studies started in 1950 with the WASH-3 report which analyzed blast damages resulting from run away reactions as potential accidents in the existing reactors. It continued with the WASH-740 in 1957 where hazards from three scenarios of fission product releases were evaluated. These studies were followed by the development of siting criteria with the 10 CFR part 100 and by the development of the Environmental Impact Statement in continuation of the National Environmental Policy Act in 1969. The well known report WASH 1400 issued in the mid 70s was the first attempt of a Probabilistic Risk Assessment (PRA) approach which will be later extensively used. It was followed by the issue in 1991 of the as well known NUREG 1150 which evaluated severe accidents risks in five US nuclear plants.

At the same time, some events in the "nuclear life" occurred which influenced significantly the safety approach. Among these events the following were recalled:

- EBR 1 incident in 1955, SL1 accident in 1961, Browns Ferry and Salem events in the 80s which led progressively to the definition of a policy for reactivity accident.
- SEMISCALE test on ECCS in 1971 which prompted the conservative approach of 10 CFR 50 and Appendix K.
- The fire which occurred in Browns Ferry in 1975 and which initiated the discussion on fire safety requirements.
- TMI2 accident which caused a redistribution in safety analysis between large break LOCA design accidents which were before too much emphasized and beyond design basis events which were neglected.

In this historical context, USNRC defined and used some principles and methods for the evaluation of plant design. A review of them has been provided which included for example conservative deterministic engineering, defense in depth, analysis of the effectiveness of maintenance (introduced in 1991). From this review it came out that there was a need for regulators to accurately predict the expected plant behavior and that this requires consequently a good understanding of the physical phenomena and processes.

The second item developed in the paper was the development of the Commission's Safety Goals and the Severe Accident Policy Statement. The first issue of these two Policy Statements was done in 1983. It was followed by a second issue, in 1985, for the Severe Accident Policy and in
1986, for the Safety Goal. The role of Probabilistic Risk Assessment (PRA) in these statements has been discussed in detail and was shown as increasing with time.

As a logical result, on the basis of all the data and knowledge gained on risk, USNRC developed a risk information decision methodology. The different steps for introducing these risk informed processes have been discussed and some examples of applications have been detailed, such as the review of the next generation nuclear power plants, Advanced Boiling Water Reactor (ABWR) from GE, 80+ from ABB, AP 600 from Westinghouse.

These regulatory activities were supported by extensive research programs. The current related activities have been discussed. The main directions of research which have been reviewed covered the CAMP effort on thermalhydraulic codes, research on aging (vessel integrity, non destructive examination,...), steam generator integrity, containment structures testing, severe accident research, advanced thermalhydraulic codes development, high burnup fuel related research, digital instrumentation and control in plants, human performance.

In conclusion, this paper provided a broad and detailed view of the evolution which occurred in the regulatory practices in US. These practices which have been more or less followed by several countries can be considered as representative of the international environment where the subject of this seminar i.e. the use of Best Estimate methods in Thermalhydraulics Safety Analysis has been developed. The three following papers of session 1 gave indeed three examples in three countries (Canada, France and Germany) on how these developments have been and are being performed.

In Canada, the situation exhibits a developing role of best estimate (realistic) calculations, complemented with uncertainty evaluations. It has been explained that this role has raised from two main reasons. First, there are some unresolved safety issues with highly conservative methods for low frequency events where BE methods can be an alternative. Second, in some cases, it has been experienced that key phenomena, plant uncertainties and modeling uncertainties had proved to be far more important than originally estimated in the analyses, even though these analyses appeared to be very conservative. This is clearly not satisfying. Two trial applications of BE calculations have been presented, one on large LOCA on Bruce B reactors, the second on loss of flow events on Darlington reactors. The problems and lessons learnt from these trials have been discussed. They highlighted some key issues: a) potential complexity of the methodology, b) adequacy of the analytical techniques and code validation, c) combination of uncertainties, d) need for stronger tie between plant operation and analysis, e) degree of statistical rigor required, f) need for good quality documentation. In conclusion, realistic calculations are considered in Canada as useful potential alternatives or additions to old conservative methods, but more work appears required both on the regulator side for developing standards and on the research side for the development of realistic analysis method.

The next presentation showed that there was also in France a clear and increasing tendency in the licensing process, to use Best Estimate codes in the analysis of safety cases. In the overall safety evaluation which includes in France the analysis of the specific assumptions, the use of the code itself and the comparison with the safety criteria, some specific cases have been analyzed using the French Best Estimate code CATHARE. These cases have been described. They cover mainly some small and intermediate break studies and analyses of emergency procedures. The best estimate approach will certainly be generalized for the future reactor EPR. A more systematic approach for licensing will have then to be defined. In support to these trends in the safety analysis area, several research programs have been started in France. Presentation of these programs was given especially on the general studies of uncertainties evaluation.
methods (IPSN approach), the development of statistical tools, the evaluation of the uncertainties of elementary individual physical models. It came out that there was still a need in getting results from the research programs in order to perform complete practical application, even if several limited attempts in real cases have been performed. Consequently, it has been recognized that the use of Best Estimate methods in licensing in France was in an intermediate state and that this situation will certainly evolve greatly in the next years.

The last paper described the situation in Germany. Here, the practice in licensing is based on deterministic thermal hydraulic analyses which have to follow rules released by the federal ministry in charge of the safety of nuclear installations. In these rules, the guidelines concerning the assumptions for the calculations are providing some latitude, and especially it is accepted that alternative assumptions, models and correlations can be used if one can prove that safety is assured at least in an equivalent way. Therefore, from the beginning, German licensing practice contained simultaneously conservative and best estimate features. Due to the increasing availability of experimental data and due to the progress in modeling, more and more realistic calculations have been replacing the conservative approaches. These calculations should be supported mainly by an improved comprehensive validation of the codes. In parallel, methods for the quantification of uncertainties of calculated results were developed but are not yet introduced in the regulatory process. For the future reactors (EPR) recommendations have been stated, for the safety demonstration, in direction of the use of realistic assumptions and models. For Germany the prerequisite for such use will be a high standard of code validation, qualified users and a reliable quantification of uncertainties. It was explained that such quantification were under development both by the designer who is developing a method similar to CSAU, and by GRS which is completing a statistical method.

In conclusion of session 1, it was clear that the papers presented were certainly not covering the complete problem but that they represented a good introduction for the presentations which had to be given next during the several sessions of the seminar. In particular the historical view of regulatory practices which was presented allowed to get a perspective of future regulations and of the role of Best Estimate approaches. The three examples of attempts in using “Best Estimate methods” in licensing which were presented raised a number of questions which were all further discussed in more detail during the seminar. The general discussion after the session reflected well this introductory character. All the items discussed, as it has been confirmed later during the seminar, converged on the fact that progress was being made but that further work was still needed to finalize the practical application of "Best Estimate Methods" in licensing.

SESSION 2:
Overview of past and current CSNI TH activities

The first paper in this session was given by Mr. Réocreux and described the "Activities of Principal Working Group 2 on Coolant System Behaviour". First, the mandate of PWG2 was described and put into context with the mission of the other working groups (Nos. 1, 3, 4 and 5). PWG2 focuses on in-vessel thermal-hydraulics during transients and accidents, core physics, fuel behaviour and core degradation during severe accidents. The main activities of PWG2 fall into four categories: 1) Providing a forum for technical exchange; 2) Review and synthesis of ongoing research (e.g. state-of-the-art reports); 3) Technical analysis (e.g. international standard problems and benchmarks); 4) Investigation of new areas. It was stressed that perhaps the most significant contributions have been in the ISPs and the development of the validation matrices. In particular, the ISPs provide a well documented (and archived) test that is invaluable
for code development and assessment and lead to the identification of the importance of the user effects.

The second paper was given by Mr. N. Aksan and discussed “Best Estimate Codes and Use of ISPs in OECD Countries”. To date, there have been 41 ISPs and two more are being prepared (Boron dilution at the University of Maryland and Passive Cooling in the PANDA facility at PSI, Switzerland). The general objectives of the ISPs are to aid in the development of codes, documenting and archiving experiments of general interest, and improve code assessment. The use of BE codes is motivated by the needs to address scaling issues (requires real description of behaviour), to quantify conservatism, and determine real plant response for EOPs. The various types of ISPs were reviewed and their benefits to the host countries, the participants and to research managers described.

The third paper was presented by both Mr. N. Aksan and Mr. H. Glaeser and gave an “Overview of CSNI, SET and ITF Validation Matrices”. Originally, PWG2 focused on the development of the integral test facility matrix assuming that separate effects assessment would be adequately covered by the code development teams. When it became clear that this was not the case, PWG2 undertook the development of a SET verification matrix to: 1) Enable continuous comparison with SET data for developmental assessment; 2) Aid in uncertainty quantification; and 3) Address scaling issues. A description of the SET and ITF matrices was then given, showing the phenomena-facility cross-reference tables and examples of the facility information available in the CSNI reports.

The last paper in the session was given by Mr. J. Misak and discussed “Thermal-Hydraulic Activities in the IAEA”. The main emphasis was on the IAEA’s efforts to provide licensing guidelines for VVERs, RBMKs and to countries in the process of developing their licensing methodologies. The most discussion resulted from the proposal to use best-estimate codes for conservative calculations. In effect, abandoning the use of evaluation models and using BE codes with “conservative” input (e.g. initial conditions, availability or capability of safety systems, etc.). This proposal evoked much commentary and the session naturally evolved into the discussion period. On the subject of using BE codes for conservative calculations, the opinions of the attendees were divided. Some strong objections were raised that this violated the philosophy of BE codes and that it would be difficult to decide how and to what extent one should maintain conservatism. Also, the question was raised concerning combinations of conservatism in a non-linear system. On the other hand, as was later evidenced in several presentations, both vendors, utilities and research institutes are pursuing conservative BE approaches. This group maintained that the use of realistic physical models was better than the Evaluation Model (EM) approach and provided an intelligent way to investigate conservatisms without the burden of performing a full CSAU-type uncertainty study.

As the discussion progressed, Mr. F. Eltawila redirected by posing the questions: ‘what are the T/H issues we face today? and what should be the CSNI PWG2 role in the future?”

It was generally agreed that phenomena identification was essentially done but that assessment was incomplete, that is, no code can adequately model all expected phenomena. For a list of code requirements it was suggested to refer to the summary of the Annapolis Meeting (1996).
SESSION 3
Part 1: BE methodologies and associated uncertainties        Chairmen: Aksan / Glaeser

UK (Nuclear Electric, AEAT, NNC)

Status:
A licensing case was performed for the Sizewell Safety Case during the 80ies. Considered were bounding limiting design basis faults (less than 100 cases). Westinghouse conservative codes (e.g. LOFT-5 code) were used and some own developments were applied. DNB failures in frequent faults were investigated, statistical 95%/95% statements derived, and additional margins applied.

Future:
A need for „better estimate“ thermal hydraulic codes was stated.

Germany, Siemens KWU

Status:
An application of a best estimate analysis plus uncertainty evaluation according to the Code Scaling Applicability, and Uncertainty Evaluation Methodology (CSAU) was presented. The bounding scenario (large break in the cold leg between pump and reactor vessel) was determined deterministically by sensitivity calculations. A statistical treatment was performed to evaluate the uncertainties of code, plant parameters and fuel. This is the first uncertainty evaluation in a licensing case for a reactor under construction.

Hungary, KFKI

Status:
In the frame of the AGNES project different initiating events are considered: DBA, PTS, ATWS. Pessimistic assumptions are applied to bound uncertainties from code model imperfections. Pessimistic moderator density reactivity coefficients were assumed. In addition, a loss of AC power at the occurrence of high cladding temperature was considered. After a review of these conservatism, the reactivity feedback was replaced by 3D reactor physics, the cladding gap conductance, and the engineering factor for the hot assembly was revised.

Future:
The USNRC and IAEA Guidelines for best estimate calculations plus uncertainty evaluation will be applied. However, no test data are available for large breaks in VVER reactors. An uncertainty analysis will be performed using the ATHLET code and the GRS uncertainty method for a pressuriser surge line break on the PMK test facility.

Canada, AECL

The CANDU reactor analysis was performed by conservative bounding in the past, and will move to best estimate analysis in the future. The best estimate codes and modules for such an analysis were presented. The treatment of uncertainties was topic of a later presentation in this Seminar (CSAU demonstration for the blowdown phase of a large LOCA on a CANDU reactor).

Spain, IBERDROLA

The utility perspective was presented.
**Status:**
Best estimate codes have been used since 1986. An intermediate approach between conservative Appendix K rules and uncertainty evaluation was discussed and applied for BWR transients. Best estimate codes are used, and $2\sigma$ values for some dominant variables and/or additive terms are applied. Additional conservatism is introduced to cover variables not considered. For LOCA analyses in BWRs a relaxed Appendix K approach was applied.

**Future:**
For BWR LOCA analyses a CSAU based method will be applied, for transients a statistical method will be used.
PWR LOCA analyses will be performed using the intermediate approach, for transients the TRACTEBEL method (a bounding method) will be used, based on the intermediate approach.

Japan, TOSHIBA
Several performed BWR analyses were presented:
- Stability
- Abnormal transient
- Reactivity initiated transient.

**Status:**
Best estimate codes are used to support the licensing analyses.

**Future:**
Aiming to use best estimate codes to upgrade the licensing analyses in the near future.

**SESSION 3 - Part 2:**
**BE Methodologies and Associated Uncertainties**
**Chairmen:** Krishnan / Toth

Papers presented in this part of the session covered topics ranging from a description and comparison of uncertainty assessment (UA) methodologies to the application of BE methods to APWR, EPR, and containment analysis. The main points that emerged in the presentations can be summarized as follows:

- Some countries are doing BE analysis on a voluntary basis.
- BE methods have been applied in EPR’s ECC system design optimization.
- BE methods have been applied to containment safety analysis.
- UA is a complex task. However, it is needed because approximations are made at every step of a BE analysis. It is therefore important to quantify the error or uncertainty in code calculation results.
- UA is necessary if useful conclusions are to be obtained from BE calculations.
- A structured approach is important in the integration of UA in code, representation and plant.

The following summarizes the highlights of the discussions and recommendations for the future:

Much progress has been made in BE analysis over the last 10 years, but it is not time yet to fully apply UA methodologies in the BE framework. Recent applications tend to quantify the uncertainties related to the most important parameters, while maintain a bounding approach for those having less influence on key parameters.
A number of rigorous UA methods are available. The challenge is to come up with a simplified method that can be used with confidence.

BE/UA should be restricted to design-basis accidents for now, until beyond-design-basis accident phenomena are well understood and codes can model them properly.

A full BE analysis may be considered in the future, but UA propagation between the various disciplines such as neutronics, thermalhydraulics, and containment must be accounted for properly.

SESSION 3: BE Methodologies and Associated Uncertainties

Conclusions

A move towards best estimate analysis in licensing is taking place. A best estimate code should predict the mean of the data (if available), rather than providing a bound to the data. The uncertainty of the calculation result in the specific application should be quantified. During this Seminar several best estimate code applications were presented but only a few uncertainty evaluations:

- 1 application of a statistical uncertainty analysis in a PWR licensing case (the first one) by a vendor (Siemens KWU, Germany);
- 1 demonstration application of the CSAU method to CANDU large break LOCA blowdown (AECL, Canada);
- 5 investigations performed in the OECD/CSNI Uncertainty Methods Study:
  - 1 extrapolation of accuracy (University of Pisa, Italy),
  - 1 bounding parameter uncertainties (AEA Technology, UK),
  - 3 statistical methods (GRS, Germany; IPSN, France; ENUSA, Spain);
- 1 investigation is under way for a PMK-2 experiment using the ATHLET computer code and the GRS Method (KFKI/ GRS, Hungary/ Germany).

Some participants prefer a bounding approach by using best estimate codes with conservative parameter values to evaluate the margin to licensing limits without a detailed uncertainty analysis. This consideration is based on their perception of cost-effectiveness. A statistical uncertainty evaluation is considered as time consuming, and a requirement to perform a statistical evaluation of uncertainties may discourage people to use best estimate codes [Y. Kukita, Japan]. However, if a licensing case is performed according to USNRC rules either a conservative analysis applying Appendix K rules has to be undertaken, or a 95% (or more) probability statement has to be provided that the licensing limits are not exceeded.

A disadvantage of applying Appendix K regulation is that the high level of code validation can not be utilised. The conservative approach is often not reflecting the physical behaviour. Also, conservative assumptions may not always lead to conservative results (for small and intermediate breaks). For example, the conservative increase of reactor power may lead to an overprediction of the swell level in the core. The resulting overprediction of cooling would be opposite to the intended conservative result.
Bounding by conservative values on a few dominant input parameters needs a demonstration of an overall conservatism on the target calculational result (maximum cladding temperature, for example). It has also to be checked that the dominant input parameters cover the contribution of other neglected input parameters. This needs to compare the results with an appropriate experimental basis and to perform sensitivity calculations.

An advantage of statistical methods is to evaluate the effect of variations of relevant parameter values and their combinations over the whole transient of an accident scenario.

The common difficulty for a bounding analysis (using best estimate codes) as well as for an uncertainty analysis is the specification of values for uncertain input parameters. This may be the bounding value, the uncertainty range or the probability distribution of an uncertain input parameter. Both, the Siemens-KWU and University of Pisa methods evaluate the code uncertainty by determining the accuracy of the calculation result compared with several similar experiments investigating the same accident scenario. Therefore, these methods need not to specify uncertain values for input parameters representing code models. Uncertainties due to scale effects are claimed to be included. The ranges or biases for uncertainties of plant conditions and fuel related parameters, however, have still to be specified (this has not yet been demonstrated for the University of Pisa Method). It was shown during the Seminar, that, in a large break scenario, the uncertainties of fuel parameters turned out to have the biggest influence.

In recent licensing cases the licensing authorities require that the uncertainty of calculation results is quantified instead of performing a conservative analysis according to Appendix K rules. It is up to the utility (or to the vendor because licensing analyses are very often contracted to the vendor) to provide an adequate evaluation of uncertainties. The first uncertainty analysis performed in the frame of a licensing case has been presented by Siemens-KWU during this Seminar.

A detailed uncertainty analysis, however, may not be necessary for results which are far from the safety limits. For these cases the distance of results from the safety limit should be clearly higher than usual uncertainty ranges, or higher than results of sensitivity calculations investigating important input parameters as well as initial and boundary conditions.

**SESSION 4:**
**Selected Issues of TH Safety Analyses**

Seven papers are part of the Session dealing with different topics or problems to be solved in relation to the application of Best Estimate Codes in the Licensing Process.

The first two papers (authors A. Tanrikut and H.R. Choi, respectively) deal with typical examples of applications of system codes to the safety analysis of NPP or to the design of new reactors. Basically, the achievement of a suitable level of maturity of the adopted codes (or code versions) is recognized: the codes can be reliably used for the fixed purposes.

Two papers in the session (presented by G. Geffraye/J. Kelly and R. Kyrki-Rajamaki: actually, the first two authors of the same paper discussed the experience gained in France and USA, respectively) deal with the problem of coupling neutronic and thermalhydraulic codes; the importance of the coupling 3-D kinetics with 1-D or 3-D system thermalhydraulics codes was outlined as well as the difficulties currently being encountered.
One paper (presented by J. Kelly) deals with the problem of applicability of current codes to new generation reactors where driving forces and velocities may be very small; consequently, some physical model (or constitutive equations) can be inadequate. The author showed, as an example, substantial improvements in predicted break flow when changing the two phase critical flow model; the applicability, in the sense of produced accuracy, of the new approach to all situations of interest may still be under discussion.

A stimulating lecture was presented by T. Speis covering an historical overview of regulatory requirements based on operational experience and occurred events. In particular, the feedback of TMI and Chernobyl accidents upon the regulatory views and the licensing needs was discussed, together with the consideration of severe accident in the PSA (Probabilistic Safety Assessment) and in the licensing process itself.

A fundamental and provocative question was put by the author, dealing with the limits of a pure theoretical PSA approach: “Can the containment system be avoided if core damage frequency is calculated to be \(10^{-7}\)?” The answer is clearly NO.

The last paper (presented by R. Kirmse) dealt with the discussion of the current status of the French-German common approach in relation to licensing. The difficulties in homogenising the interpretation of safety requirements was outlined. However, the activity, still in progress, aims at a deep re-evaluation of the safety principles with the general objective of a reduction of the core melt frequency; among the specific objectives (just examples are reported here), there are the elimination from the current PSA sequences, of accident situations which could lead to large early releases of radioactive materials, and the need of emergency evacuation from the immediate vicinity of the plant when low pressure core melt situation should occur.

**SESSION 5: Final discussion**

In this session, the discussions and reactions from the host country participants indicated that this Seminar was a very valuable and interesting meeting with a collection of papers reflecting the state of the art for the use of the best-estimate methods in thermal-hydraulics analysis and also neighboring subjects, e.g., neutronics, containment, PRA and severe accidents. They have now better appreciation of what needs to be done, if they go for nuclear power in the future. They also stressed the importance of having an access to various information and experimental data through participation in CSNI-PWG2 activities and hope to be more active in the future.

Discussions show that there is a very strong need to use better-estimate codes instead of conservative codes in licensing. Presently, some countries prefer bounding approach by using best-estimate codes with conservative parameter values, conservative boundary and initial condition assumptions to evaluate the margin to licensing limits. This is done without the use of a detailed uncertainty evaluation analysis, due to cost effectiveness considerations and time requirements for the necessary analysis. In this sense, existing uncertainty methods are found to be useful but there is a strong need to apply to different transient types. Since the uncertainty methods which have been developed are very rigorous, the challenge is to come up with simplified method which can be used with confidence in licensing process. A full best-estimate approach with uncertainty analysis may be considered in the future. In addition, there is a transfer of uncertainties when computer codes are coupled from different disciplines, e.g., thermal-hydraulics, neutronics, and containment codes. These "transfer of uncertainties" are known, but the problem is not addressed in detail, presently.
C. PAPERS

SESSION I: KEYNOTE PRESENTATIONS
NRC Regulatory Philosophy for Commercial Nuclear Power Plants

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Ankara, Turkey
June 29, 1998

I am pleased to be here today to share with you some NRC perspectives on nuclear reactor regulation.

The mission of the Nuclear Regulatory Commission is to regulate the Nation's civilian use of byproduct, source, and special nuclear materials to ensure adequate protection of the public health and safety, to promote the common defense and security, and to protect the environment. This mission was articulated by the US Congress for the NRC in the Atomic Energy Act of 1954 and the Energy Reorganization Act of 1974. The NRC has carried out its mission through the regulations that it has enacted throughout the years. The evolution of these regulatory requirements for the community of US operating nuclear power plants is the focus of this talk.

My talk is divided into four parts: first, an historical overview, including major studies of the safety of nuclear power plants and the development of power reactor design basis accidents; second, the Commission's Safety Goals, Severe Accident Policy, and the uses the staff makes of the goals and policy; third, the recent move toward more formal risk-informed decisionmaking, including a look at how Advanced Light Water Reactors have utilized this approach; and, finally, some research programs that are germane to this area.

We'll start with history. The predecessor of the NRC, the Atomic Energy Commission, undertook studies of the safety of the small critical assemblies that existed at the time and were used for research and radionuclide production. The AEC's Reactor Safeguard Committee published WASH-3 in March 1950, after 15 months of study. This document considered the existing reactors from the point of view of blast damage, from run-away reactions; release of radioactivity to the air; accidental release of stored fission product activity to water supply systems; and "other reactor hazards," which mainly consisted of radioactive iodine and argon releases from stacks under normal operating conditions. As you can imagine, the analysis methods were crude, but it is interesting that the Committee developed “general formulas and criteria which it considered were useful in evaluating hazards.” Finally, the Committee noted, in the introduction to the report, that nuclear energy, then used only for research, for production of radio-isotopes, and for production of material for weapons, was expected to be used also for commercial power production.

The next major study of reactor hazards was published in 1957, prior to the start-up, also in 1957, of the first US commercial power plant, at Shippingport, Pennsylvania. Rather than being based on an actual design, the study, WASH-740, was based on a set of rather arbitrary assumptions. The report evaluated three reactor accident scenarios: first, the contained case, where all the fission products were vaporized inside the containment shell, which was assumed to remain intact; second, the volatile release

27
case, where all the volatile fission products were assumed to be released through a
breach in the containment or a failure to secure all openings; and, third, the 50 percent
release case, where 50 percent of all fission products -- volatile and non-volatile -- were
assumed to be released. The report pointed out that substantial consequences to the
public could be expected from the more severe 2 of the 3 scenarios. Qualitative "factors
for or against" having a major accident were discussed and orders of magnitude were
given, based on expert opinion. Using the most pessimistic guess for the probability of
having an accident, which was 1 chance in 100 thousand per year, and the most
pessimistic evaluation of the number of fatalities in such an accident, which was 3
thousand, the report estimated 1 chance in 50 million per year that a person would be
killed in a reactor accident for a 100-reactor industry.

In 1962, the then-existing siting practices were documented in guidance to be used to
develop siting criteria -- that is, distance requirements -- for nuclear power plants. The
siting criteria had been codified in 10 CFR part 100, which specified that dose limits at
the site boundary and at the outer edge of the low population zone were 25 Rem to the
whole body or 300 Rem to the thyroid gland, whichever is more limiting. In practice, the
controlling limit is that to the thyroid. The guidance, which was contained in TID (for
Technical Information Document) -14844, was referenced in Part 100. TID-14844
specified that the release into the containment to be postulated for the "maximum
credible accident" was all of the noble gas, half of the radiiodine, and 1 percent of the
solid fission products. Overall, this release represented about 15 percent of the total
fission product inventory. The noble gas and iodine release are still used today for
design basis accident evaluation, but the solid fission product release, which was to be
used to evaluate "shine" from the containment, has not survived in this age of heavily
shielded containments.

A new dimension was added to reactor licensing with the passage in 1969 of the National
Environmental Policy Act, NEPA, which mandated that environmental impacts of all kinds
-- radiological, aesthetic, chemical, and a host of others -- be considered for all "major"
Federal actions. The NRC prepares an Environmental Impact Statement for each
reactor site under licensing review; early EISs considered all accidents in the design
basis, while EISs after 1980 considered, in addition, a wider range of accidents based on
probabilistic risk assessments.

In the mid-1970s, the Reactor Safety Study, often referred to as WASH-1400, was
issued. It was the first attempt to take a comprehensive look at the risk -- throughout this
talk, I will mean probability times consequences when I use the term risk -- from nuclear
power plant operation using a systematic, numerical approach applied to a specific
Pressurized Water Reactor and a specific Boiling Water Reactor. This numerical
approach was called Probabilistic Risk Assessment, or PRA. The results were
extrapolated to the same 100-plant industry as had been assumed for WASH-740. The
Commission chartered an outside review of WASH-1400 and a report was issued that
noted that the methodology was sound and an advancement over earlier methodologies
and should be developed and used more widely. However, it also noted that
WASH-1400 "leaves much to be desired" and found that the Executive Summary was a
poor description of the content of the report. In response to the criticism, the
Commission decided not to use PRA in decisionmaking, and the technique languished
for many years.

The final study I want to mention is NUREG-1150, entitled "Severe Accident Risks: An
Assessment for Five U.S. Nuclear Power Plants," which was published in January 1991.
As the name implies, five different power plants were evaluated. PRA methods were used, but a major advance was the extensive evaluation of the uncertainties. Expert elicitation methods were developed and elicitation was used to prepare the required parameter distributions. Further, where WASH-1400 relied on a series of small computer codes to evaluate fission product release and transport, NUREG-1150 used system level severe accident codes, which performed coupled analyses in the reactor coolant system and in the containment.

While all these studies were being made, the experience in the industry was also influencing the design and operation of plants. The early plants, particularly test reactors, were relatively small and remotely sited. The assurance of safety was primarily attained through their small size, through the ability of the containment building to accommodate core melting that might occur and through the low population density surrounding the sites. Contemporaneous research focused on reactivity control and reactivity insertion accidents because of the natural concern about the ability to control and bound the nuclear reaction. This concern was further reinforced by a 1955 test, which had been planned to investigate core instabilities, went amiss at the Experimental Breeder Reactor, the EBR-I. The power level was allowed to rise to very high levels and shutdown was accomplished with a slow, rather than a “scram”, mode. Partial melting of the metal alloy core ensued, but there were no off-site releases of radioactive material. Further, in 1961, an accident caused by pulling the extremely high worth central control rod -- by hand! -- at the SL-1 reactor led to three fatalities. Two events in the early 1980s, at commercial plants, prompted the NRC to focus increased attention on the Anticipated Transient Without Scram -- or ATWS -- leading to a rule on the subject in 1984. The first event was a failure in the scram discharge volume at Browns Ferry, which caused a common mode failure of control rods to insert. The second event occurred at Salem, where faulty maintenance practices caused the trip breakers to fail to open on demand. As a matter of fact, problems with control rod scram continue to this day. We have seen problems in Pressurized Water Reactors with control rods failing to insert completely in high burnup modules and problems in Boiling Water Reactors with sticking diaphragms in the scram inlet and outlet valves.

By the mid-1960s plants increased substantially in size as the designs evolved from research to proof-of-concept to commercialization. Powers increased from the order of 300MWe, at Shippingport, to 3000MWe, which is the power level we see today. At the same time, sites were selected that were closer to the electricity users, bringing higher population densities. Sites inside cities were even being considered, though they were never approved, and siting was a major issue of the day. The increasing plant size was driven by the need to achieve economies of scale in order to be able to compete commercially.

The increase in power lowered the probability that containment could accommodate postulated core melt sequences, leading to the possibility of overpressurization or basement melt-through, the latter possibility leading someone with a poor grasp of geography to coin the phrase “The China Syndrome.” It was recognized that the containment approach to assuring safety of the public needed to be reconsidered, since there was significant uncertainty as to whether containment integrity could be maintained in an uncontrolled core melt event. Inclusion of Emergency Core Cooling Systems in new and operating plants followed the 1971 SEMISCALE experiment, which had resulted in large bypass of the emergency core coolant during a test that simulated a loss of coolant accident -- or LOCA. There were protracted hearings on the subject and, in view of the relatively limited information existing in the early 1970s, conservative criteria were
adopted in January 1974 as 10 CFR 50.46 and Appendix K. The criteria mandated conservative assumptions for license applicants to follow at each step of the process in performing safety analyses. The conservative criteria were later relaxed, following major improvements in our ability to calculate the relevant phenomena. In 1975, a fire occurred at the Browns Ferry plant. It resulted from a worker looking for air leakage paths using a candle. Because the operators did not want to use water on an electrical fire, it was allowed to burn until almost all of the cabling for safety and non-safety systems was disabled. The incident was terminated -- barely -- without damage to the core or releases of radioactive material off site, but it resulted in the NRC’s subsequent attention to fire safety at all plants. That attention led to requirements that are given in Appendix R to Part 50. Fire safety requirements are a subject of discussion even today.

No review of operating experience would be complete without a discussion of the accident at Three Mile Island, Unit 2. That 1979 accident ranks as the most important failure of the regulatory and safety review process to date in the US. The large NRC thermal hydraulic research program that was underway was not organized to uncover such a design weakness. The WASH-1400 study of reactor safety published in 1975 had highlighted the need for increased attention on small break LOCAs and transients with multiple failures or operator errors because of their higher probability of occurrence. However, the design basis accident concept emphasized single failures. Actually, the implications of a precursor event at Davis Besse had been recognized by individuals at NRC and Babcock and Wilcox and the potential for such an event had been realized by an electrical engineer at the Tennessee Valley Authority. Yet the flow of information was such that the staff at TMI was never informed. This flaw in the system led to the creation at the NRC of the Office for Analysis and Evaluation of Operating Data, with the very same TVA engineer as its first Director. On the industry side, the Institute for Nuclear Power Operations was formed.

Following the TMI-2 accident, the Commission recognized that it needed to examine the state of knowledge on source terms more closely and to think beyond the design basis accident framework, in order to be able to make technically sound regulatory decisions. Early studies provided the basis for the Severe Accident Research Program. A report entitled “Reassessment of the Technical Bases for Estimating Source Terms,” issued as NUREG-0956 in 1986, reviewed the state of knowledge from that program and assessed future research needs. This information, and other related plant-specific risk studies, provided the technical basis for developing NUREG-1150.

Over some 30 or 40 years, a period punctuated by the major studies and operating experience I mentioned, the NRC developed a set of regulations and requirements built upon a defined set of accidents and hazards -- the design basis envelope. The design pressure and temperature of the containment are dictated by the evaluation of a large break LOCA, which is evaluated assuming the failure of the single, highest worth train of mitigating equipment and the concurrent loss of offsite electric power. However, the fuel temperature and the time at that temperature are constrained to be those which prevent embrittlement of the fuel rod cladding. Prevention of embrittlement is deemed to be so important because of the need to preserve a coolable geometry in the core by ensuring that the cladding will not crumble during the reflood phase of the accident. Experiments showed that cladding could remain intact if as much as 17 percent of the thickness were oxidized. Nonetheless, the hydrogen that will be produced during the oxidation phase of the accident, before reflood, is assumed to be what would result from oxidation of 100 percent of the cladding surrounding the active fuel and plants must be able to
accommodate the production of so much hydrogen by, for instance, having hydrogen recombiners or having a containment inerted with nitrogen. So the design basis LOCA must be calculated to be terminated with the fuel rods intact. Nonetheless, the containment leaktightness requirement comes from using the source term in TID-14844, which, as I mentioned before, comes from a core melt evaluation. This dichotomy of requirements remains in the design basis today.

The NRC has used the suite of design basis accidents to evaluate the combination of the designs of power plants and power plant sites using traditional engineering approaches of defense-in-depth, codes and standards, and conservative deterministic engineering. The principles developed for plant design included diversity, redundancy, defense in depth, and reliability. Emphasis was placed on systems, structures, and components -- or SSCs -- with defined safety functions and most requirements were associated with those SSCs. Although event likelihood and consequences were considered qualitatively in selecting the design basis envelope, no formal risk assessment tools were applied to reactor safety within the design basis. In the latter 1960s, these design principles were codified in the General Design Criteria, given in Appendix A to Part 50. A lot of attention was paid to accident prevention in these criteria, but some attention was paid to mitigative systems -- an example of the former is the Emergency Core Cooling System or ECCS and an example of the latter is the containment. The defense-in-depth philosophy emerged as a four phased approach. Three of the phases are related to ensuring the integrity of physical barriers: first, the cladding of the fuel rods; second, the pressure boundary of the reactor coolant system; and third, the high pressure containment building. The fourth barrier to exposure of the public following an accident is of a different character: the emergency planning that is done around each of the plant sites.

In 1991, the Commission added a new dimension to the regulatory framework. Requirements for monitoring the effectiveness of maintenance at nuclear power plants was added to the regulations as section 50.65. This regulation is performance-based, as we call it, in that results are prescribed, not processes, and goal setting, monitoring, and trending are required to ensure compliance. The expected result is reasonable assurance that systems will perform their intended functions by avoidance of maintenance-preventible functional failures of SSCs that are within the scope of the rule. The goals for what constitutes that reasonable assurance are established by each individual licensee. The industry was given 5 years to prepare itself for full implementation of the rule, that is, until July of 1996. Since then, the NRC has been inspecting implementation processes and procedures established by each licensee. One of the major considerations has been the enforcement posture to take in the face of deficiencies in implementation of a performance-based rule. In the early inspections of the implementation processes, enforcement has been taken for both process and performance deficiencies; later, the NRC will be taking action only for failure to use the maintenance rule to adequately address SSC performance problems.

The initial designs of US commercial reactors included significant margins of safety for a variety of vital parameters. Explicit margins were established and required by the various design codes and regulations. They have evolved over time as operational experience has accumulated and design analysis methods have improved. Implicit margins were also included. These margins resulted from the conservative assumptions and materials data that were incorporated into design, procurement, construction, and operational guidance.
In granting operating licenses to power plants, a judgement is made by the reviewers at the NRC that there is reasonable assurance that the plant can be operated without undue risk to the health and safety of the public. The NRC must assure that nuclear power plants are designed, constructed, and operated in a manner that is consistent with this objective.

The agency operates according to the principle that safety of plant design, construction and operation is the responsibility of the licensee. The agency must, however, have the capability to independently assess plant designs and safety analyses submitted by license applicants and to review the general implications resulting from operating experience. Regulators must have the capability to accurately predict expected plant behavior during all normal and upset conditions so that regulatory actions affecting design, operation, and maintenance can be made on a firm technical basis. In practice, this requires the ability to understand and describe the physical phenomena and processes that may occur.

Turning now to the development of the Commission’s Safety Goals and the Severe Accident Policy Statement.

In 1983, the Commission issued Policy Statements on both Safety Goals and Severe Accidents. Numerous comments were received on both statements and they were significantly revised. I will discuss each of the revised Policy Statements in turn.

The Severe Accident Policy Statement was reissued in 1985 to describe the policy the Commission intended to used to resolve safety issues related to reactor accidents more severe than the design basis. Further, the Statement focused on the criteria and procedures the Commission intended to use to certify new designs for nuclear power plants. Briefly, for operating reactors, the Commission announced that it would formulate an approach for searching for possible significant risk contributors -- called outliers or vulnerabilities -- that might be missed absent a systematic search. This program came to be known as the IPE, or the Individual Plant Examination, and includes a search for severe accident vulnerabilities due to events initiated by both internal and external events. Also, as part of this policy, the staff undertook the Containment Performance Improvement Program to use surrogate plants of each containment type to search for generic improvements that could be evaluated for application to each of the plants in the class. Also, the staff has been considering accident management strategies for classes of plants and classes of accidents that could improve guidance to plant management in case of a severe accident. Relative to next-generation reactors, the Commission stated that it fully expected that vendors engaged in designing new standard (or custom) plants would achieve a higher standard of severe accident safety performance than their prior designs.

The Safety Goal Policy Statement, as it was reissued in 1986, put forth two governing principles:

First, individual members of the public should be provided a level of protection from the consequences of nuclear power plant operation such that individuals bear no significant risk to life and health; and, second, societal risks to life and health from nuclear power plant operation should be comparable to or less than the risks of generating electricity by viable competing technologies and should not be a significant addition to other societal risks.
To translate these into something measurable, two objectives, called the Quantitative Health Objectives -- or QHOs -- were established.

First, the risk to an average individual in the vicinity of a nuclear power plant of prompt fatalities that might result from reactor accidents should not exceed 1 tenth of 1 percent of the sum of prompt fatality risks resulting from other accidents to which members of the U.S. population are generally exposed; and, second, the risk to the population in the area near a nuclear power plant of cancer fatalities that might result from nuclear power plant operation should not exceed 1 tenth of 1 percent of the sum of cancer fatality risks resulting from all other causes.

A subsidiary objective was given that the probability of core damage accidents should be less than 10-4 per reactor year. This was not an objective equal to the QHOs at the time, and still is not, but the Commission has requested the staff to consider if it should be elevated to the status of a goal and the staff owes the Commission a recommendation on the subject.

In publishing the Policy Statement for comment, the Commission requested comments on whether there should be a goal on containment performance. The comments were negative on the subject and the Statement as issued in final form did not include such a goal. However, the Commission requested the staff to explore the usefulness of defining a large release of radioactive material that could be associated with a desired probability of 10-6. The staff concluded that it was not possible to define a release that would not be more restrictive than the goals the Commission had already established and the Commission instructed the staff to discontinue its efforts. However, the NRC is still concerned about containment performance as well as about core damage frequency, and I will be talking about both of those again later.

Following the issue of the Safety Goal Policy Statement, the Commission turned more to probabilistic safety studies, performed utilizing computer codes and the physical understanding developed through research. Such studies provide a method of cataloging and arranging in order of significance the accident sequences representing serious threats to the fuel, the reactor coolant system, or the containment.

The use of probabilistic risk assessment and the evaluation of operating experience led to research into events that tend to dominate risk. Transients to be considered may be revealed by PRA, suggested on the basis of operating experience, or, simply, postulated. Operational transients normally have consequences within the design basis envelope, but may warrant special consideration if they are seen as precursors to more serious events. The use of PRA and operating experience provided new insights into previously unconsidered systems interactions, common mode failures, multiple failures, or operational errors.

Gradually, the structure and discipline of probabilistic methods have led to developments which for the first time permits logical appraisal of uncertainties and of deficiencies in our understanding. For example, for NUREG-1150, detailed containment event tree and uncertainty analyses were performed and the probability of core damage, containment failure, and offsite consequences were presented in terms of a range of values within which the true value would likely reside rather than as just a single point estimate of the mean. These ranges evolved from explicit consideration of uncertainties. The technical basis for assessing risk is now on a much sounder footing. Having said that, I quickly have to point out that model and completeness issues still remain, not only in areas we
already include, such as human performance, but also in areas we don’t yet have a firm method for dealing with, such as fire, shutdown, and low power risks.

During the last several years, both the NRC and the nuclear industry have recognized that PRA has evolved to the point where it can be used increasingly as a tool in regulatory decisionmaking. In August 1995, the NRC adopted a policy statement regarding the expanded use of PRA that has four major conclusions: First, the use of PRA technology should be increased in all regulatory matters to the extent supported by the state-of-the-art in PRA methods and data and in a manner that complements the NRC’s deterministic approach and supports the NRC’s traditional defense-in-depth philosophy.

Second, PRA and associated analyses, for example, sensitivity studies, uncertainty analyses, and importance measures, should be used in regulatory matters, where practical within the bounds of the state of-the-art. The objective is to reduce unnecessary conservatism associated with current regulatory requirements, regulatory guides, license commitments, and staff practices. Where appropriate, PRA should be used to support the proposal of additional regulatory requirements in accordance with 10 CFR 50.109, the NRC’s Backfit Rule. Appropriate procedures for including PRA in the process for changing regulatory requirements should be developed and followed. It is, of course, understood that the intent of this policy is that existing rules and regulations shall be complied with unless these rules and regulations are revised.

Third, PRA evaluations in support of regulatory decisions should be as realistic as practicable and appropriate supporting data should be publicly available for review.

And, fourth, the Commission's safety goals for nuclear power plants and subsidiary numerical objectives are to be used with appropriate consideration of uncertainties in making regulatory judgments on the need for proposing and back fitting new generic requirements on nuclear power plant licensees.

In its approval of the policy statement, the Commission articulated its expectation that implementation of the policy statement would improve the regulatory process in three areas: first and foremost, through safety decisionmaking enhanced by the use of PRA insights; second, through more efficient use of agency resources; and, third, through a reduction in unnecessary burdens on licensees.

In parallel with the publication of the policy statement, the NRC staff developed an implementation plan to define and organize the PRA-related activities being undertaken. These activities cover a wide range of PRA applications and involve the use of a variety of PRA methods (with variety including both types of models used and the detail of modeling needed). The activities described in the PRA Implementation Plan relate to a number of agency interactions with the regulated industry. With respect to reactor regulation, activities include, for example, guidance development for NRC inspectors on focusing inspection resources on risk- important equipment, and a reassessment of plants with relatively high core damage frequencies for possible mandated changes to plant designs, the process we call “backfitting.”

My third topic of discussion is risk informed decisionmaking. As a result of the extensive body of risk information available, it became clear that such information can provide very useful insights to improve our regulations and practices. In fact, some of the NRC's generic actions, such as the ATWS rule and the station blackout rule, had been
supported by risk assessments as part of the “backfitting” process. In addition, these
insights have identified other areas important to risk that may be over- or
under-regulated. Therefore, to focus attention on those items of most importance to
public health and safety, the Commission is turning to explicit use of risk informed
decisionmaking.

The Commission recently approved Regulatory Guide 1.174 and the corresponding
Standard Review Plan chapter covering the general guidelines for use of risk information
in requests to change plant licensing bases. Those documents provide that proposed
changes are expected to meet a set of five key principles. These principles are:

First, the proposed change meets the current regulations unless it is explicitly related to
a requested exemption or rule change.

Second, the proposed change is consistent with the defense-in-depth philosophy.

Third, the proposed change maintains sufficient safety margins.

Fourth, when proposed changes result in an increase in core damage frequency or risk,
the increases should be small and consistent with the intent of the Commission’s Safety
Goal Policy Statement.

And, fifth, the impact of the proposed change should be monitored using performance
measurement strategies.

Each of these principles should be considered in a risk-informed, integrated
decisionmaking process.

The proposed evaluation approach and acceptance guidelines follow from these
principles. In implementing these principles, it is expected that all safety impacts of the
proposed change will be evaluated in an integrated manner as part of an overall risk
management approach. The approach should ensure that the licensee will make broad
use of risk analysis to improve operational and engineering decisions. For those cases
where risk increases are proposed, the benefits should be described and should be
commensurate with the proposed risk increases. The approach used to identify changes
in requirements should be used to identify areas where requirements should be
increased, as well as where they could be reduced.

The guidelines for review and approval of proposed changes define 3 areas in the map of
core damage frequency versus change in core damage frequency and in the map of
large early release frequency versus change in large early release frequency. The core
damage frequency is the same subsidiary objective I spoke about before in connection
with the Safety Goal Policy Statement. I mentioned then that there was no goal on
containment performance, although the NRC continues to be concerned about the
subject. To deal with this issue, the concept of “large early release frequency” was
developed to be a surrogate for the early fatality quantitative health objective. It is defined
as the frequency of those accidents leading to significant, unmitigated releases from
containment before effective evacuation of the close-in population, such that there is a
potential for early health effects. The areas in both parameter spaces define where the
increases in risk posed by the changes are deemed to be very small so that only a
normal review would be necessary; where the NRC management will have to give
attention to the changes in risk, though the changes are still small; and where no
changes would normally be allowed. These areas are bounded by "fuzzy" -- rather than sharp -- lines of demarcation. But, roughly, they are: for very small changes, a little above 10-4 in core damage frequency and up to 10-6 in change in CDF and, for small changes, up to 10-4 in CDF and up to 10-5 in change in CDF. No changes would be allowed for values larger that those. The boundaries of the corresponding areas in large early release frequency are a factor of ten lower in every case.

To enlarge on the general guidance on the use of risk information in the regulatory arena, the staff has prepared specific guidance for four applications. They are: improved Technical Specifications, inservice testing, graded quality assurance, and inservice inspection of piping. The regulatory guidance and standard review plans -- or inspection guidance as the case may be -- for the first three have been sent for approval to the Commission following resolution of public comments. The last, inservice inspection, is behind the others in schedule, but should be ready for issue in the fall of this year.

Risk evaluation has been an integral part of the review of the next-generation nuclear power plants. The NRC has been actively reviewing three applications for design certification of new designs; they are: the Advanced Boiling Water Reactor from General Electric, the System 80+ from ABB-Combustion Engineering, and the AP600 from Westinghouse. The design certifications have been issued for the first two and the Final Design Approval is expected for the AP600 by the end of this year. The regulations in Part 52 concerning standard designs require a probabilistic risk assessment and all three of these designs were reviewed for severe accident capability. This review consisted of a traditional engineering review of the severe accident design features and a review of the probabilistic assessment. I will discuss each of these review aspects.

The goal of the traditional engineering review was to resolve issues associated with: first, operating experience, including TMI-2; second, the results of PRA that were performed on new designs and current designs; third, early efforts on severe accident rulemaking; and, fourth, research conducted to address previously identified generic safety issues. It was considered desirable to resolve the issues with design features and not to rely on analyses. The ABWR and the System 80+ were reviewed against standards developed by the staff. As I mentioned earlier, the Commission had determined that the next-generation designs should achieve a higher level of safety than current plants and the staff did conclude in Safety Evaluations of these two designs that this had been achieved. As I mentioned, the staff has not completed its review of the AP600, so no Safety Evaluation of this design has been issued.

The probabilistic assessments of the three designs were, for the first time, available early in the development of the designs when modifications could be most effectively implemented. While there were some differences in what was actually done for each design, the analyses considered internal and external events, included sensitivity and importance studies, and considered shutdown events. The designs were subjected to traditional PRA analysis for some parts of the analysis, for instance internal events, but utilized other, simplified, probabilistic techniques for other aspects, such as the Seismic Margins Method for seismic analysis. The AP600 design includes no active safety-related systems, relying on passive features for accomplishing safety functions within the design basis. These passive design features allow long times to be available before operator action is required. However, the design includes active non-safety-related systems, whose function it is to reduce challenges to the passive systems. The probabilistic assessment for this design had as one task to support the regulatory treatment of these non-safety-related systems. For all three designs, the
insights developed during the design-specific PRA review resulted in changes to the
designs to reduce risk and also identified important information to be considered during
design, construction, testing, and operation of the facilities.

I’ll turn now to my last topic, some research programs that are germane to this area. If I
were to review for you the history of NRC’s research accomplishments, the rest of my
talk would be as long as the talk has been so far. So I will focus on our current research
programs. The US, as has been true in most other countries, has budget pressures that
have caused us to seek out cooperative research with many other countries. As some
of you know, Turkey is a partner with the NRC and other countries in the Code
Application and Maintenance Program -- the CAMP effort. We have had cooperative
international programs for many years, but, recently, we have redoubled our efforts to
stretch our research dollars by working with others. Several of the programs I will be
discussing are funded in this cooperative manner.

The NRC has had programs related to aging research since the 1960s. The problem of
a single reactor vessel and the issue of neutron embrittlement led to some of this early
research. Research continues today in this same area. Our aging program is focused
on the need to provide the data and analysis tools necessary to identify, quantify,
manage, and regulate the effects of aging in nuclear power plants. Aging affects virtually
all active and passive components in a nuclear reactor system. It stems from exposure
to reactor operating temperatures, irradiation environments, both primary and secondary
water environments, cyclic operation -- that is, fatigue -- and general wear. However, the
specific aging-related degradation mechanisms can be difficult to identify, and their
effects may be difficult to quantify. In addition to the research on vessel integrity, the
NRC has an active program in non-destructive examination -- NDE -- techniques and
analytical evaluations. Advanced inspection systems are evaluated by experienced
inspectors and insights about potential pitfalls and reliability of the equipment are
identified.

Steam generator integrity has been a issue for many years. The industry continues to
experience about one steam generator tube rupture every two years, and tube
degradation from several mechanisms has continued to plague many plants. The work
in this area involves developing independent data, analyses, inspection methods,
predictive models and acceptance criteria for evaluating steam generator tube integrity.

Over the years several small- and medium-scale models of containment structures have
been tested to failure. The latest of these was in 1996, when a steel Boiling Water
Reactor containment model was tested. Analytical efforts are still underway and a
review of the post-test analyses will lead to an assessment of our analytical capabilities.
A pre-stressed concrete model is being built and test plans are underway.

In response to Congressional direction, the NRC reduced its severe accident research
program. However, we continue to pursue advanced thermal-hydraulic code
development efforts and high burnup fuel issues. The objective of the code development
effort is to complete and validate one thermal-hydraulic code to replace several that the
NRC has been maintaining. We expect to exploit new technology to make the code work
better and faster and to be less expensive to maintain and to use. We will improve the
database structure and incorporate improved two-phase flow modeling.

Relative to high burnup fuel issues, the NRC is obtaining data from hot cell tests and
from test reactor programs in several countries, including Norway, France, Japan, and
Russia. We intend to update our fuels computer programs and to analyze relevant situations to determine their safety significance.

Obsolescence is forcing power plant operators to move from analog instrumentation to digital. Numerous issues have arisen ranging from how are such systems to be validated to how will they respond to environmental stress. New methods for evaluating software quality will be tested at the Halden Reactor Project in Norway. We are also developing review guidance for digital instrumentation and controls systems under various conditions of temperature, humidity, electromagnetic interference, and smoke.

We have known for many years that human performance is a major contributor to the risk profile at a plant -- operators can be excellent first- and last-lines of defense or they can be the cause of a problem by doing, or not doing, something of importance. We do not yet know how to accurately model human performance in PRAs and we don't have the information necessary to assess the impact of operator performance on nuclear safety. Nor can we assess the influence of such issues as human-system interfaces, procedures, training, and organizational factors. We have programs in these areas, but the issues are difficult. A final issue relative to human performance is the effect that deregulation of the electric power industry in the US may have on operator and plant performance. We know that safety and economic operation can, and often do, go hand-in-hand. However, we continue to watch developments as deregulation comes and plant owners have to compete in an open marketplace. We remain concerned about the stability of the grid under deregulated conditions and about operations at some of our poorer performers.

In summary, I have tried to trace the development of NRC's regulatory philosophy for commercial nuclear power plants in a way that describes the development in terms of studies and events that shaped its character. The future of regulation will be determined by developments in the risk-informed arena and the deregulated environment.
NRC Regulatory Philosophy for Commercial Nuclear Power Plants

Ashok C. Thadani
Acting Deputy Executive Director for Regulatory Effectiveness

June 29, 1998 -- Ankara, Turkey
HISTORY

Studies

- Technical Information Document - 14844
- National Environmental Policy Act
- Reactor Safety Study - WASH-1400
- NUREG-1150

Events

- Browns Ferry Fire
- Three Mile Island Unit 2

Regulations

- Design Basis Accident
- General Design Criteria
- Maintenance Rule
SAFETY GOALS AND SEVERE ACCIDENT POLICY

Severe Accident Policy

- Individual Plant Examination
- Containment Performance Improvement
- Accident Management
- Next-Generation Designs

Safety Goals

- Goals
- Quantitative Health Objectives
- Subsidiary Objective

PRA Policy Statement

- Increase Use of PRA
- Use for Regulatory Matters
- Realistic Evaluations
- Use Safety Goals in Judgments
RISK-INFORMED DECISIONMAKING

General Regulatory Guide

- Meet Current Regulations
- Defense-in-depth Philosophy
- Safety Margins
- Small Change
- Monitoring

Application Specific

- Technical Specifications
- Inservice Testing
- Graded Quality Assurance
- Inservice Inspection of Piping

Next-Generation Designs

- Advanced Boiling Water Reactor
- System 80+
- AP600
RESEARCH PROGRAMS

- International Cooperation
- Aging
  - Non-Destructive Examination
  - Steam Generator Integrity
- Containment
- Thermal-Hydraulic Code Development
- High Burnup Fuel
- Digital Instrumentation
- Human Performance
Regulatory Framework Concept

Adequate Protection Region
(Cost Not a Consideration)

Cost-Beneficial Region
(Cost is a consideration)

Safe Enough Region

Additional regulation may be warranted

Additional regulation not warranted
Acceptance Guidelines for Core Damage Frequency (CDF)

Region I
- No Changes Allowed
- Track Cumulative Impacts

Region II
- Small Changes
- More Flexibility with respect to baseline CDF
- Track Cumulative Impacts

Region III
- Very Small Changes
Acceptance Guidelines for Large Early Release Frequency (LERF)

Region 1
- No Changes Allowed
Region II
- Small Changes
- Track Cumulative Impacts
Region III
- Very Small Changes
- More Flexibility with Respect to Baseline LERF
- Track Cumulative Impacts
The Developing Roles of "Best-estimate" Thermal-Hydraulic Calculations and Uncertainty Analyses in Licensing in Canada

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OECD/CNSI Seminar on
Best-Estimate Methods in
Thermal Hydraulic Safety Analysis

Ankara, Turkey
29 June to 1 July 1998
The Developing Roles of "Best-estimate" Thermal-Hydraulic Calculations and Uncertainty Analyses in Licensing in Canada

Abstract

This paper discusses, from a regulatory perspective, the developing roles of "best-estimate" (or realistic) thermal-hydraulic calculations and uncertainty analysis in making licensing decisions in Canada. Realistic calculations can be useful for a variety of reasons. Examples are to avoid unnecessary economic penalties, remove overly restrictive operational practices or improve operational flexibility, deal with plant ageing effects, and help resolve outstanding safety analysis issues. Uncertainty assessments are a necessary complement to realistic calculations, providing information on the sensitivity of analysis results to modelling and/or plant variations.

In Canada, the developing role of these types of analyses has arisen for two main reasons. First, for some low frequency events which traditionally have been analysed with extreme conservatism, there are safety analysis issues which are not being resolved as rapidly as the regulator would like, and which therefore warrant an alternative approach. Second, experience over recent years has shown that even though licensing analyses appeared to be very conservative, for some events, key phenomena and plant and modelling uncertainties had proved to be far more important than originally estimated. These experiences have prompted a) the regulator to request more systematic and comprehensive approaches to assessing plant and modelling uncertainties and b) the industry to start developing methods to take full advantage of best-estimate codes.

Two specific trial applications of realistic calculations are focussed upon: large LOCA for the Bruce B reactors; and loss of flow events for the Darlington reactors. These trials are being used by the regulator to assess the feasibility of moving towards a more methodical use of realistic calculations in licensing. The problems encountered and lessons learnt from these trials are discussed; some of the key issues are a) the potential complexity of the analysis methodology, b) adequacy of the underlying analytical techniques and computer code validation, c) the method by which uncertainties are combined, d) the potential need for a stronger tie between plant operation and the analysis (operational compliance), e) the degree of statistical rigour required, and f) the need for good quality documentation. Parallels are drawn between the experience gained in Canada and those of other countries.

In conclusion, the trial applications have shown that realistic calculations and uncertainty analyses are useful alternatives/additions to more conservative methods. In Canada, for such methods to become an established part of the licensing framework, further work is required by the regulator to develop standards, and by the licensees to fully develop their realistic analysis methods.
1.0 Introduction and Background

In Canada, nuclear energy is regulated by a federal agency, the Atomic Energy Control Board (AECB). The current Atomic Energy Control Act is to be replaced by a new Nuclear Safety and Control Act, likely to come into force this year\(^1\). The new Act will be supported by revised regulations and complemented by a set of revised regulatory policies, standards and guides. The content of the regulatory guides will be based to a great extent on currently existing guides. In revising the guides for safety analysis, the AECB Staff is taking the opportunity to make clearer its expectations\(^2\) with respect to analysis rules, computer codes and analysis acceptance criteria. In re-evaluating the safety analysis rules, the AECB Staff is also assessing the feasibility of allowing licensees to use best-estimate\(^3\) methods for licensing analysis. This paper discusses this developing role of best-estimate calculations and the factors that both the licensees and regulator must consider for these methods to become established as licensing analysis methods.

*Canadian Regulatory Philosophy*

The Canadian regulatory philosophy is that the licensee has primary responsibility for safety and that detailed regulatory prescription is unnecessary and detrimental to the licensee carrying out that responsibility. The AECB Staff sets high-level safety goals and standards, audits licensees' performance against these and enforces compliance with the regulations. The interpretation of the high level requirements is left to the licensee. This leads to a licensing process with relatively little formal regulatory prescription, and, to a very large extent, licensees are free to choose the methods by which safety is demonstrated.

This minimal prescription is particularly evident for analysis methods and acceptance criteria; there are no "Appendix K" type analysis rules, and few acceptance criteria are prescribed by the regulator. In demonstrating that the reactor design meets the high-level design criteria, a licensee is free to choose the analysis methods, computer codes and quantitative derived acceptance criteria which it considers to be appropriate. Therefore, there has been much flexibility for Canadian licensees to pursue best-estimate computer codes and analysis methods.

*Derived Acceptance Criteria*

The AECB Staff sets two types of analysis success criteria: radiological dose limits and high level qualitative design criteria. The dose limits are prescribed in regulatory guide C-6 [1] (shown in Table 1). The dose limits are graduated according to the frequency of the event, i.e. lower limits for more frequent events.

\(^1\)The AECB will become the Canadian Nuclear Safety Commission.

\(^2\)The intent is to codify accepted AECB Staff practices, rather than introduce new regulatory requirements.

\(^3\)In this paper the term "best-estimate" is used to indicate that the techniques attempt to predict a realistic plant response. Best-estimate is not used in the statistical sense (USNRC regulatory guide 1.157).
Table 1
Radiological Dose Limits

<table>
<thead>
<tr>
<th>Class</th>
<th>Example of event</th>
<th>Effective dose (mSv)</th>
<th>Eye Lens (mSv)</th>
<th>Skin (1 cm²) (mSv)</th>
</tr>
</thead>
<tbody>
<tr>
<td>1</td>
<td>SG tube rupture</td>
<td>0.5</td>
<td>5</td>
<td>20</td>
</tr>
<tr>
<td>2</td>
<td>Small LOCA</td>
<td>5</td>
<td>50</td>
<td>200</td>
</tr>
<tr>
<td>3</td>
<td>Large LOCA</td>
<td>30</td>
<td>300</td>
<td>1200</td>
</tr>
<tr>
<td>4</td>
<td>class 1 + no ECCS</td>
<td>100</td>
<td>1000</td>
<td>4000</td>
</tr>
<tr>
<td>5</td>
<td>class 2 or 3 + no ECCS</td>
<td>250</td>
<td>1500</td>
<td>5000</td>
</tr>
</tbody>
</table>

Licensees must also show that the reactor design meets high level qualitative design criteria. These criteria are based upon past experience and historical precedent, and are codified into regulatory guides (for example [2,3,4]). To demonstrate compliance with a qualitative design criterion, it is transformed into a set of quantitative criteria known as "derived acceptance criteria". Responsibility for the development of these lower level criteria has been traditionally placed on the licensees and designers, not the regulator.

As an example, for LOCA events, an important high-level design criterion is that fuel channel integrity is maintained [3,4]. To meet this qualitative criterion important derived acceptance criteria to be considered are:

- limit on fuel sheath temperature;
- limit on fuel centre-line temperature;
- limit on fuel string constrained expansion;
- limit on fuel sheath to pressure tube contact;
- limit on pressure tube strain at high pressure.

The specific limits which a particular licensee chooses to support vary.

Frequently, the emphasis of the regulatory review is placed upon whether the analysis meets the derived acceptance criteria, rather than the dose limits, because the former tend to be more difficult to meet.

**CANDU Reactor**

The CANDU is a pressurised, heavy water moderated, and heavy water cooled, channel reactor. The fuel is natural uranium which is typically contained within 37 element fuel bundles. The reactor core consists of a lattice of horizontal fuel channels (typically 380) encased in a calandria vessel which contains the moderator. A closed loop heat transport system (HTS) is provided to transfer the heat from the fuel to the secondary side light water steam. Important safety systems that may be used to mitigate the consequences of an accident or event are the "special safety systems" (shutdown systems SDS 1 and SDS 2, containment and emergency core cooling (ECC) systems); standby emergency systems (which include emergency electrical power, boiler emergency cooling and emergency service water); and process systems (for example, reactor regulating
system, boiler auxiliary feedwater, primary circuit feed and bleed, and normal electrical power).

From a thermal-hydraulic safety analysis perspective, the CANDU reactor has some distinctive features and characteristics:

- the fuel resides in a matrix of individual horizontal fuel channels;
- the primary circuit (heat transport system) is relatively complicated;
- the moderator system is separate from the coolant and is at low pressure;
- re-fuelling is performed at power;
- the reactor has a positive core void reactivity coefficient.

These features are significant because some result in relatively tolerant analysis acceptance criteria, whilst others place certain requirements on the safety analysis methods. Because the fuel is contained within individual channels, rather than in an open lattice, concerns regarding the exothermic effects of fuel sheath oxidation are considerably reduced. The requirement to demonstrate fuel channel integrity results in a derived acceptance criterion for sheath temperature, but a low limit (such as 1204 °C) is neither necessary nor imposed. With respect to analysis methods, the above features can result in a complex safety analysis process. For example, for large LOCA, there is a need to integrate the result of a number of reactor physics, thermal-hydraulic and fuel codes.

2.0 Confidence in Safety Analysis

In the event of a plant transient or accident a high level of confidence is needed that the plant's safety systems will perform as intended. This level of confidence must be reflected in the licensing safety analysis. It is derived, to a large extent, from:

- conservative initial plant operating states, which are chosen to pessimise the consequences with respect to the analysis success criterion under consideration;
- conservatively crediting only specific mitigating systems and equipment, taking into account active component single failures, and the minimum allowable performance characteristics;
- conservative modelling assumptions;
- validated computer codes and analysis methods; and
- margins to established derived acceptance criteria.

These individual aspects are discussed in turn below.

*Initial Plant Operating State: "Limit of the Operating Envelope" Assumption*

Currently, many licensing safety analyses for CANDU reactors are performed with the assumption that all relevant plant operating parameters are simultaneously at their worst permissible values; for many parameters these values correspond to license limits. The analysis is intended to maximise the consequences under consideration and thus be bounding for all possible plant operating states. This is referred to as "limit

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4This operating envelope includes all possible plant states including highly transient ones that may last for a few minutes during power manoeuvres. An exception is the analysis of over power protection for fuel channel integrity which credits the low probability of refuelling ripple in the initial power distributions.
of the operating envelope” (or LOE) analysis. For the majority of the time the plants operate well away from these limits, so that the analyses done at the LOE predicts more severe consequences than would be expected for more probable (and realistic) plant states.

The probability of LOE plant states is usually extremely low. For example, for a loss of flow event the LOE plant state has the reactor power, channel power and maximum bundle powers all at their licensing limits, and the flux tilt, reactor inlet header temperature and reactor outlet header pressure at their operational limits. The probability of operating with such a combination is extremely low. Although such an operating condition is highly improbable, this does not necessarily imply that similar consequences would not be obtained by analysing a more probable plant state.

For many events LOE analysis is a straightforward, (relatively) inexpensive and effective approach. However, there are instances where the LOE approach can create regulatory discomfort. For example, for the more severe consequences associated with some LOE analysis, representative experimental support of the analysis methods can be difficult to obtain. These concerns will be sensitised when safety margins are small or the accident consequences are excessive.

Mitigating Systems and Equipment Credits

Safety analyses are generally performed to demonstrate the capability of special safety systems to meet their performance requirements. Active component single failures are included and limiting performance characteristics are assumed. For example, in evaluating SDS 1 performance, the two rods with the highest reactivity worth are assumed unavailable and maximum rod drop times are assumed.

The extent to which process systems (such as the reactor regulating system (RRS), HTS feed and bleed system, normal electrical systems, secondary side control systems, etc) are credited depends on the type of special safety system under consideration. For example, in evaluating shutdown system performance, the RRS is not credited, and for many events, other process systems are either too slow to respond or not relevant. In contrast, in evaluating ECC or containment systems’ performance, process systems are frequently credited. The credited performance of the process systems is usually best-estimate, although allowance may be made for unavailability of certain equipment.

The performance of the shutdown systems is made particularly demanding by requiring that each SDS is analysed independent of the other SDS and of the other special safety systems.

Conservatism in Computer Codes and Modelling Assumptions

Historically, the amount of conservatism built into different analytical models and computer codes has varied considerably. For example, reactor physics and fuel codes have generally been designed as best-estimate, some of the older generation of thermal-hydraulic codes were designed to be very conservative, and the newer generation of thermal-hydraulic codes have been designed as best-estimate. In addition, the predicted consequences are frequently sensitive to specific models or assumptions which are subject to uncertainty. Where this is the case, these models or assumptions have been modified to make the analysis demonstrably conservative. In some situations, highly stylised modelling assumptions are also used to simplify the analysis methods. Consequently, in applying a suite of codes there can be a mix of best-estimate and conservative codes, models and modelling assumptions.
Developing Role of Best-Estimate Analysis

Such an approach can lead to difficulties in that:

- with a mix of assumptions with differing conservatism, the overall degree of conservatism in the analysis is unknown; although this in itself is not necessarily a problem, when small margins exist to analysis acceptance criteria, questions arise over the trade-off between conservative and best-estimate assumptions;

- where highly stylised modelling assumptions are used, the conservatism can be excessive, leading to predicted consequences which are undesirable and experimentally unsupportable5; such assumptions also mask the "true" margins to acceptance criteria.

There is an increasing reliance on the use of best-estimate codes for licensing analysis. With such codes there is a corresponding need to ensure that they are adequately validated and that modelling uncertainties are taken into account into the accident analysis. The licensees have adopted a systematic process to validate their codes, but, other than for a few isolated cases, the issue of modelling uncertainties has not been addressed systematically. For those cases where specific models have been made conservative, this has been done on an ad hoc basis and other important modelling uncertainties have not been identified or taken into account.

Computer Code Validation

Confidence in the computer codes used to predict the response of a plant to an event comes from a good understanding of the underlying physical processes and phenomena, their interactions, and the correct modelling of those phenomena in the computer codes (through fundamental equations, correlations, constitutive relationships, etc). A particularly important way of demonstrating that a code has the required predictive capability is through validation, the quality of which is highly dependent upon the quality of the experimental data. Test facilities can be categorised into two types for validation purposes6. Separate Effects Test (SET) facilities are used to validate the ability of the code to predict specific phenomena or small groups of phenomena. Integral Effects Test (IET) facilities contain all the main components of a nuclear plant and are intended to simulate all the main phenomena and their interactions.

One of the key factors in the adequacy of the SETs and IETs data is the relevance of the geometry and experimental conditions to the plant event for which validation is required. It is necessary that the test facility exhibit all the important phenomena expected to occur in the plant, and that the test conditions are sufficiently close to predicted plant conditions to avoid the need for gross extrapolation of models and correlations beyond their validated ranges.

Achieving representative conditions in the SETs is particularly important, because these tests provide the basic foundation for the code validation. For some events and phenomena, achieving conditions fully typical

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5As an example, for a number of years Ontario Hydro supported "limit consequence" analysis for large LOCA. Such analyses made highly stylised assumptions about the effectiveness of the ECCS. The analysis was considered to be conservative, but it resulted in extreme conditions and phenomena that could not be supported by experimental data.

6Data from plant transients and accidents is usually of limited use due to the minimal instrumentation and sampling frequencies.
of the plant can be difficult and/or prohibitively expensive. Also, in some instances, it is not clear that the SETs include all of the relevant phenomena anticipated to exist in the plant events. Difficulties in achieving plant representative conditions are due to avoiding incurring significant damage to the test facility; power limitations; geometric scaling compromises; and the use of materials and fluids with atypical physical properties (including the limited ability to consider affects of irradiation).

The IETs suffer the same compromises described above for the SETs, but to a much greater extent. The conditions for these tests can be much less severe than for the corresponding plant analysis, and the geometric scaling compromises more pronounced than for SET facilities. Also, IETs tend to be less well instrumented, and there can be concerns over instrumentation accuracy. Therefore, any conclusions drawn from these tests, or from any code validation, have to be extrapolated to the plant and this can be done only with limited confidence.

In practical terms, the SET and IET experimental database will never be fully representative of plant conditions, and codes will always be used as extrapolative tools rather than interpolative ones. In particular, achieving data for thermal-hydraulic conditions relevant to LOE analysis may be either not possible or cost prohibitive. The question is: how can the lower confidence associated with extrapolative tools be increased? One way is through best-estimate analysis.

3.0 The Need for Best-estimate Analysis and Uncertainty Analysis

The above discussion highlights a number of reasons why currently accepted analysis approaches can lead to a lack of confidence in the analysis.

- For the more severe consequences associated with some LOE analysis, representative experimental support of the analysis methods can be difficult to obtain.
- The use of overly conservative assumptions can result in severe predicted consequences for which experimental support is difficult to obtain.
- For a number of accident sequences the SET and IET experimental database is not fully representative of plant conditions.
- Codes must be used as extrapolative tools rather than interpolative ones.
- When reliance is placed upon best-estimate computer codes, a more systematic approach to modelling uncertainties is required.

Overall confidence in safety analysis comes from a number of elements. It is important that the correct balance of confidence is achieved amongst the elements. Excessive conservatism and confidence in one element can lead to reduced confidence in another element and reduced confidence overall. For example, analysing very conservative initial operating states or using highly conservative modelling assumptions provides high confidence in one sense, but results in lower confidence if the predicted consequences are beyond the bounds of experimental knowledge, or the margins to acceptance criteria are very small.

In Canada, the developing role of these types of analyses has arisen for three main reasons. First, to help redress the regulator’s lack of confidence (described above) in some analyses. Second, for some low

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7For example, pressure tube ballooning phenomenon during a CANDU large LOCA; the accumulator nitrogen blowdown phenomenon in a PWR large LOCA.
frequency events which traditionally have been analysed with extreme conservatism, there are safety analysis issues which are not being resolved as rapidly as the AECL Staff would like, and which therefore deserve an alternative approach. Third, experience over recent years has shown that even though licensing analyses appeared to be very conservative, for some events, key phenomena and plant and modelling uncertainties had proved to be far more important than originally estimated.

There are a number of additional factors which are driving both the AECL Staff and industry into considering best-estimate calculations.

- Licensees have, on occasion, experienced substantial operating penalties and reduced operational flexibility. To help avoid the risk of future penalties and alleviate operational constraints, licensees are developing analysis methods to utilise their best-estimate codes to increase or restore licensing safety margins.
- For specific events there has been diminishing safety margins as the licensees improve their understanding of the phenomena and incorporate this into the analysis. There have been instances where conservative methods resulted in minimal licensing safety margins and regulatory discomfort. Best-estimate calculations provide a means of demonstrating the more likely safety margin.
- Ontario Hydro has recently recognised (based on findings from their Independent Integrated Performance Assessment) that the safe operating envelope is not as well defined as originally thought; improvements in its definition may require a best-estimate analysis approach.
- There are emerging safety issues and station ageing concerns that may be best resolved with best-estimate analyses.

It is also recognised that best-estimate analyses have additional advantages; they can

- provide information on how the plant would behave in response to a real accident, which is required for input to emergency operating procedures for correct event diagnosis by the operators;
- provide information on riskier plant states for the operator to avoid; and
- provide more realistic consequence analysis in support of probabilistic safety assessments.

4.0 Trial Applications of Best-estimate Calculations and Uncertainty Analyses

4.1 Bruce B Large LOCA

Background

Following the discovery of a previously unrecognised phenomenon that affects the magnitude of the power pulse, Bruce B units had been running at reduced capacity. To support operation of the Bruce B units at higher power, analysis of the power pulse phase of the large LOCA event was required. This particular analysis was somewhat different from previous ones in that it included a best-estimate calculation and also an assessment of modelling uncertainties for certain important parameters. These analyses were done in addition to the usual licensing analyses.
Thermal-hydraulic Overview of the CANDU Large LOCA Event

The large LOCA consists of five accident phases: an initial power pulse phase (0-5 seconds), an early blowdown phase during which channel deformation can occur (typically 5-50 seconds); a late blowdown phase during which the ECCS starts to inject and quench the fuel (typically 50-150 seconds); a refill phase; and a long term ECCS recirculation phase. From the perspective of fuel channel integrity (the focus of this discussion), the important phases are the first two.

The power pulse phase is characterised by an initial rapid depressurisation of the heat transport system, voiding of the core, a positive insertion of reactivity due to the positive core void coefficient, and a power excursion as energy is rapidly deposited into the fuel. The excursion is terminated by the action of the two shutdown systems. As a consequence of the energy deposition and reduced cooling, there is a substantial increase in fuel temperatures; typical licensing analysis peak temperatures are 2000 °C at the centreline and 1300 °C on the sheath. In this phase of the accident the other key parameter is fuel string expansion relative to the expansion of the corresponding pressure tube.

During the second accident phase, the high fuel sheath temperatures can impart sufficient energy to the pressure tubes to cause these to become ductile. Since the HTS pressure is still high the pressure tubes strain radially into contact with their respective calandria tubes. Once in contact, heat can be deposited into the moderator system via the calandria tube. Typical licensing analyses predict that of the order of 100 pressure tubes will strain.

The pressure tube straining process has been examined through a substantial experimental programme, and the conditions under which pressure tubes strain without rupturing is thought to be understood. Nevertheless, there have been some outstanding technical concerns and the straining process is less well characterised under "outlier" behaviour or conditions. When such a large number of pressure tubes are predicted to balloon, outlier behaviour becomes increasingly important in demonstrating that the consequential failure of a single fuel channel will not occur. From a regulatory perspective this is undesirable.

So, for this particular analysis the key parameters of interest were those important to demonstrating fuel channel integrity: fuel centre-line temperature, fuel sheath temperature, fuel string axial expansion, and the number of pressure tubes ballooned.

Best-Estimate Analysis

For the best-estimate analysis, the key operating parameters that were used as inputs to the computer codes were set at 90th percentile limits rather than at limit of the operating envelope values. The basis for the particular choice of parameters was that they were monitored, could be controlled, and were important in determining the consequences of the accident. The parameters were power distribution, shutdown system performance, coolant and moderator isotopic purities, moderator poison, and ECCS performance. The selection of values for use in the analysis was based on statistical analysis of historical operational data. Engineering judgment was used to determine best-estimate values for key parameters that could not be

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8Consider the LWR analogy of fuel clad ballooning. When the effect is restricted to a small number of rods there is little concern over the potential for a localised blockage. When the number of rods is large, outlier behaviour becomes more important and the limited experimental data may not capture all the relevant phenomena. This was a significant concern for the UK regulator, HMI, in its assessment of the safety case for the Sizewell 'B' PWR.
analysed statistically. The analysis credited both shutdown systems and a single failure rule was not imposed. Also, allowance for modelling uncertainties was excluded.

The best-estimate analysis predicted maximum values for fuel centre-line and sheath temperatures approximately 400 °C below those of the equivalent licensing analysis calculation. With respect to the relative fuel string axial expansion, the licensing analysis predicted that the fuel was close to compression, whereas for the best-estimate analysis the expansion was reduced by almost 50%. Finally, the best-estimate analysis predicted that a few or no pressure tubes would balloon into contact with their respective calandria tubes, representing a significant improvement over the large number of pressure tubes predicted to balloon in the licensing analysis.

This particular best-estimate calculation was not submitted as, nor intended to be, a rigorously defendable licensing calculation. However, it clearly demonstrated the potentially large margins achievable with more realistic calculations.

Uncertainty Analysis

For the licensing analysis, an assessment of modelling uncertainties was performed for three consequence parameters: fuel centre-line temperature, fuel sheath temperature and fuel string axial expansion. Note that the licensing analysis was done at the LOE and that the uncertainties were applied to this calculation rather than to the best-estimate calculation. This is consistent with the approaches used for the Sizewell ‘B’ large LOCA uncertainty analysis [5] and the USNRC’s relaxed interpretation of Appendix K prior to amending the regulations to permit full best-estimate models [6].

The uncertainty methodology consisted of five basic steps:

- identify and rank possible sources of modelling uncertainty;
- for those models ranked high, determine the range of uncertainty;
- for the limiting break location, perform sensitivity plant calculations for each of the highly ranked models;
- derive a combined uncertainty, by summing in quadrature, the results of the individual uncertainty calculations; and
- perform additional sensitivity analysis to support the use of summing in quadrature.

The identification and ranking process was performed for all reactor physics, thermal-hydraulics and fuel models and parameters9 and each model or parameter was ranked as high, medium or low. Models that were ranked high included decay heat, depressurisation rate, fuel sheath-to-coolant heat transfer, UO2 thermal conductivity and UO2 thermal expansion. Many of these are model uncertainties considered in LWR analyses (for example [5,7,8]).

The assessment assumed that the individual uncertainty contributors were independent and that the combined uncertainty could be conservatively estimated by adding in quadrature the individual sensitivities.

In comparison with the licensing base case the uncertainty analysis showed increments of approximately

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9Uncertainty in the void reactivity coefficient was excluded from the ranking because the analysis used a regulatory mandated value.
220 °C on fuel centre-line temperature;
170 °C on fuel sheath temperature;
3 mm on fuel string axial expansion.

Overall, the modelling uncertainty assessment showed that the effect of modelling uncertainties on the predicted consequences can be significant, and can be as important as uncertainties arising from initial plant state or safety system performance.

4.2 Darlington Loss of Flow

Background

As part of Ontario Hydro’s programme to update the Safety Report, a new safety analysis methodology, called the Operational Parameter Methodology (OPM), was being developed. This analysis methodology was being used on a trial basis for loss of flow events and loss of moderator/cooling events, and regulatory concurrence on the analysis methodology had been sought. In 1997, it was discovered that the trip coverage for loss of flow events at Darlington was substantially worse than that previously stated to exist. Darlington operated at 55% full power until analysis was submitted to show that the design and operational changes would result in adequate trip coverage at high power. The analysis methodology used was based on the OPM.

Thermal-hydraulic Overview of the CANDU Loss of Flow Event

The particular event of interest is one in which power is lost to a single heat transport system (HTS) pump. Following the pump trip the reactor regulating system (RRS) would normally stepback the power to 0.5%; however licensing analysis does not credit this process. The rundown of the pump causes a decrease in the coolant flow-rates in the affected HTS loop and a mis-match between the heat generation in the fuel and the heat removed. The reduction in flow results in an increase in coolant temperature and the appearance of void toward the channel outlets. There is a corresponding rise in HTS pressure, and redistribution of coolant between the two HTS loops and into the pressuriser. Due to the voiding in the channels there is a rise in neutronic power, and this is partially compensated by action of the RRS. Depending on the precise event sequence and initial conditions, reactor trip signals may be initiated by high HTS pressure, SDS instrumented channel low flows or HTS low flow.

The two principal goals of the analysis are to show that the trip parameters are robust in tripping the reactor and that fuel failures do not occur. Regulatory guidance on analysis acceptance criteria for this event specifies prevention of the onset of intermittent fuel sheath dryout for the primary trip parameter.

Best-Estimate Analysis

The focus of the analysis methodology was on assessing the uncertainty in operational parameters and the impact of these on the effectiveness of the trip parameters. The proposed analysis methodology consisted of the following features.

Identify sensitive plant operational parameters. This is done through sensitivity analyses.
Analyse plant operation. Statistical distributions of the sensitive parameters are determined from plant operational data.

Define "Operating centre", "worst normal" and "worst analysed" values for the sensitive parameters. An operating centre value for a system parameter is the observed mean plus an allowance for the mean ageing. The worst normal value for a plant parameter is equal to the operating centre plus tolerances for variation of processes and system parameters, and random and systematic instrument uncertainties. Simulation error is included if relevant. The worst normal value is representative of a 90th percentile value at a 90% confidence level. Worst analysed values are defined in a similar way to worst normal, but the allowances for process variation and instrument uncertainties are assessed at a higher percentile and to a higher confidence level. The worst analysed value is representative of a 99.9th percentile value at a 95% confidence level.

Operating centre calculation. This analysis is done with all operational parameters set to the operating centre values. This analysis demonstrates the most probable safety margin.

Worst normal calculation. This analysis is done with all operational parameters simultaneously set to their worst normal values. If the number of sensitive parameters is large this may represent a low probability plant state.

Worst analysed calculations. A number of worst analysed cases are done with all parameters but one set to their worst normal values and with the remaining parameter set to its worst analysed value. These cases are used to define the licensing limits for the operational parameters and to help define the safe operating envelope. For each operational parameter identified as sensitive an action limit will be defined. The purpose of these action limits is to signal to the operator that action must be taken to prevent the parameter from exceeding the safety analysis value or licensing limit and to ensure that the historical statistics of the parameters can be used to reliably predict their future behaviour.

Analysis acceptance criteria. Different analysis acceptance criteria are proposed for worst normal and worst analysed analyses; less demanding criteria are set for the worst analysed cases because of their lower probability.

Operational compliance. A key element of the OPM is operational compliance. In order that operation of the plant is consistent with the analysis assumptions, operational compliance strategies are required for the sensitive operational parameters. Compliance strategies help ensure that

- the trip parameter setpoints assumed in the analysis remain valid;
- certain important plant parameters remain within operational limits; and
- the historical statistical distributions of the important plant parameters continue to be representative of future operation.

The strategy for compliance is based primarily on three principles: prevention, detection and response. Prevention involves the recognition of events which could lead to sensitive parameters exceeding desired values and developing and implementing procedures and processes to minimize the probability of these events from occurring. Detection involves monitoring sensitive parameters and alerting the operator to evolving conditions affecting sensitive parameters. Response involves the development of procedures which mitigate the extent and duration of operation with sensitive parameters at undesirable conditions. Some
compliance strategies are "hard", requiring operational monitoring and operator intervention when OP&P limits are approached and/or exceeded. These are imposed through operator alarm limits and action limits. Other compliance strategies are "soft", simply requiring off-line monitoring by shift engineers.

4.3 Lessons Learnt and Problems Encountered in the Trial Applications

In the regulatory assessment of these analyses the AECB Staff reached a number of conclusions about the usefulness of best-estimate calculations in making licensing decisions. The analyses also highlighted some of the potential difficulties.

*Increased confidence in analysis.* The more extreme conditions predicted in LOE analysis can result in a considerable gap between experimental and predicted in-reactor conditions. The Bruce B trial application demonstrated that the thermal-hydraulic conditions predicted in the best-estimate calculation were much closer to the conditions in validation experiments than for the equivalent LOE analyses. For the best-estimate analyses there is likely to be a much higher degree of confidence in the code validation. For the Darlington application the improvement is less clear (see "Larger margins" below).

*Greater likelihood of resolution of some long-standing safety issues.* The current approach to safety analysis in Canada has been to perform conservative/bounding analyses and then use experimental evidence to show that the analyses are truly conservative. This approach has failed to solve certain issues for two reasons; firstly, the analyses predict outcomes which are undesirable, and secondly, it is difficult to support these outcomes experimentally. For the issue of pressure tube ballooning, the Bruce B best-estimate analysis demonstrated a possible alternative path to resolution.

Examples of where a best-estimate plus uncertainty analysis approach could potentially provide benefits are: pressure tube ballooning in a large LOCA, molten fuel/moderator interaction for single channel events, steam-generator tube degradation, and SDS 1 shutdown depth for single channel events.

*Larger margins to analysis acceptance criteria.* The Bruce B trial application demonstrates significant improvement in margins for fuel channel integrity. For the Darlington loss of flow application, the advantage is less clear: the "operating centre" calculation shows significant margins, whereas the "worst normal", "worst analysed" and LOE calculations predict similar consequences. (One possible reason is that the probability associated with the worst normal calculation is still very low; alternatively, it may be that there is much less advantage with best-estimate methods for loss of flow events.)

*Modelling uncertainties must be systematically addressed.* The Bruce B uncertainty analysis clearly shows the importance of modelling uncertainties for large LOCA. One of the AECB Staff criticisms of the Darlington application was the relatively small effort applied in examining uncertainties in the models. Not only must the examination be systematic, but the process must be scrutinizable by a regulator or peer reviewer. This is particularly important when engineering judgement or expert opinion is used (for example, in identifying and ranking phenomena and models). The Bruce B parameter ranking process raised questions: how to deal with problems which involve a number of different technical disciplines (is everyone's interpretation of "high" the same?); what is an acceptable approach if a large number of medium ranked phenomena are identified?

*How important is the method for combining uncertainties?* The method for combining the uncertainties in
the Bruce B application was quite crude. Despite this, the additional sensitivity calculations confirmed that for this particular application the approach was likely conservative. Clearly, such a method will not be applicable in all cases, particularly if there are strong inter-dependencies between models with high uncertainty. In the Darlington application, the effect of the operational parameter uncertainties was compounded, likely an unnecessarily conservative approach. However, a method should not be condemned purely on the basis of its simplicity, provided margins to the acceptance criteria are sufficiently large to accommodate the additional conservatism.

Statistical analysis of station operation. The Darlington analysis raised a number of issues over the way in which the station operational data should be analysed. For example, distinctions should be made between: deliberate operating states and stochastic ones; different plant operating configurations. The manipulation of the plant data (for example, distribution non-normality, pooling of data from all units, data averaging) must be statistically sound.

Complexity. An inherent difficulty with best-estimate analyses are their complexity. This is a feature of any methodology which does not analyse all potentially limiting plant states. This complexity places additional demands on the QA aspects and in particular on the interfaces between individual technical disciplines and between analysis, and operational organisational units.

Cost. Both trial applications (Darlington in particular) showed that to develop a sound best-estimate analysis methodology is costly. In addition, because of the increased complexity, there is a substantially higher cost of regulatory review. The increased complexity emphasises the need for good documentation and effective dialogue between the developer and regulator or peer reviewer.

4.4 International Experience

There has now been a number of applications of best-estimate calculations and best-estimate analysis methodologies in the international community. The OECD and CSNI have helped to promote such methods through workshops, publications and seminars such as these. However, many of the methods have not developed beyond the R&D phase, and in terms of licensing for design basis accident analysis, very few methodologies have either sought or gained regulatory acceptance; fewer still have been publicised.

The USNRC sponsored the development and demonstration of a method that would meet the intent of the revised Federal Regulations, permitting the use of best-estimate calculations in licensing analyses. The initial application was to a large LOCA in a 4 loop PWR and the methodology is described in detail in [8] and has come under scrutiny through international peer review [9]. It has since been applied to a number of different events (see for example the overview in [5]). The significance of the method is that it provides a very general framework which is applicable to a wide range of situations, including different events and reactor types. In the US it has become the de-facto standard for best-estimate analysis methods.

Westinghouse has developed a best-estimate analysis methodology applicable to large LOCA for Westinghouse 4 loop PWRs. It is the only best-estimate method to have undergone regulatory review and approval in the US. In reviewing the methodology, the Advisory Committee on Reactor Safeguards indicated the importance of the CSAU initiative [10]. Thus, the final Westinghouse methodology mirrors the CSAU.
5.0 Considerations for Utilising Best-estimate Analysis in Licensing

In considering a more prominent role for best-estimate calculations in licensing safety analysis, there are a number of issues. Many of the issues became apparent in the Staff's review of the trial applications. Some of these are technical whilst others relate to regulatory philosophy. At this time we do not have all the answers. The AECB Staff intends to develop regulatory guidance on how licensees should approach these issues.

5.1 General

Overcoming the philosophy that conservatism is always good. From a regulatory perspective there can be a temptation to think that conservatism is always a good feature in an analysis. Yes, but only up to a point. If extreme conservatism in an analysis leads to station operational inflexibility or burdensome operator practices which diminish safety, or harsh economic penalties, or poor confidence in the analysis, then the need for that conservatism must be scrutinised carefully.

Excessive conservatism, such as highly stylised modelling assumptions, can also mask potential safety issues. For example, the possibility of a core re-heat following the quench in a PWR best-estimate LBLOCA [11] would not have been identified as a potential issue if the best-estimate nature of reflood and quench had not been investigated (i.e. if sole reliance had been placed in the Appendix K stylised reflood behaviour). A balance in the level of conservatism must be achieved.

There is a small probability that the station can operate in an unanalysed state, or that the analysis acceptance criteria may be exceeded. Although best-estimate analyses represents a departure from what traditionally has been accepted in the past, the approach is consistent with a move towards risk-informed regulation. The key is to know how small is the probability, what the consequences would be, and whether there may be an associated cliff-edge. Although these improbable combinations of accidents and plant states would not be analysed as part of the licensing design basis, the associated residual risk from such combinations could nevertheless be assessed, through PSA and severe accident programmes, and be shown to be acceptably small.

Analysis options open to the licensee. Traditionally, AECB Staff have not prescribed specific analysis methods. Consistent with this practice, the choice of analysis methodology (i.e. LOE or best-estimate) would remain with the licensee. (This is similar to the amended US regulations which permit the licensee to choose a traditional Appendix K method or a best-estimate one.)

However, whichever approach is chosen AECB Staff expects the analysis methods and codes to be supportable by experimental evidence to a reasonable degree of confidence. Generally, the expected degree of confidence is higher for events with higher probability. For some events, this degree of confidence may not be achievable for LOE plant conditions and for these cases the licensee will be expected to provide an alternative means of resolution. One acceptable means of resolution would be through the use of best-estimate analysis.

For events in which there are large margins to the acceptance criteria and the analysis is well supported by experimental evidence, there may be little benefit in employing a best-estimate approach.

Maintaining consistency with past practice, regulatory philosophy and policy. It is important that a best
estimate approach be broadly consistent with past practice and that it is a logical and reasonable progression thereof. Consistency with past practice can be ensured by retaining certain deterministic rules. Examples of such rules are:

- analysis must be done to demonstrate the effectiveness of each shutdown system, independent of the other (R-8);
- limiting failure of a single component of a shutdown system must be assumed (R-8);
- limiting failure of a single component of an ECC system must be assumed (R-9).

Best-estimate analysis will be permitted in revision 1 of C-6, but will not necessarily be compliant with other current consultative and regulatory policy documents, for example [2,3,4]. Any acceptable best-estimate analysis approach would need to take into account, to some extent, the requirements in these documents, and modification to some of these documents will be required. For example, clarification would be needed as to how the single failure rule is applied in the assessment of special safety system performance and how minimum allowable performance standards relate to the best-estimate analysis assumptions.

**Increased cost associated with best-estimate analysis.** The up-front development costs associated with best-estimate analysis methods is likely high. However, once the methodology is established, the costs should be similar to current analysis methods. Best-estimate methods also require an assessment of uncertainties; however, since licensees are increasingly employing best-estimate codes, these are needed anyway. Best-estimate methods may also require additional plant parameter surveillance and compliance activities; the cost of these activities should be relatively small once set up. Because best-estimate methods are more complex and may result in tighter links between the analysis and plant operation, the cost of regulatory review will be higher.

**Improved basis for the derived acceptance criteria.** Traditionally, the derived acceptance criteria have been developed by the licensees and regulatory approval sought. There has been a tendency for the licensees to push the criteria as far out as possible to maximise the analysis margins and, in doing so, the criteria have encroached on certain high level design criteria. As a consequence, there has been disagreements between AECB Staff and the licensees on some of these criteria. Although, this issue is not specific to best-estimate analysis, agreement on these criteria will be required before licensees can take full advantage of best-estimate methods.

### 5.2 Technical Issues

There are a number of general technical issues which will need to be considered before a best-estimate analysis method gains regulatory approval.

**Use of deterministic assumptions.** Traditionally, deterministic assumptions have been used in analysis typically to allow for operational flexibility, account for modelling uncertainties, maintain the claimed independence between special safety systems and process systems, or introduce a degree of conservatism to simplify the analysis. Certain of these deterministic assumptions have become firmly established as part of the design and licensing basis for the system concerned (for example, unavailability of shutoff rods with the highest worth; crediting only one shutdown system). These deterministic assumptions need to be identified, and the extent to which they can be relaxed should be determined. We expect the deterministic assumptions to be chosen such that the intent of C6, R7, R8, R9 and R10 would be met.
For LWRs a similar approach was used by the USNRC in amending the requirements of § 50.46 and Appendix K in the Federal Regulations. Analyses must employ the single failure rule, and consider the loss of off-site power (see Appendix A to 10 CFR Part 50).

What probability level, percentile, and confidence level? Currently, there is no accepted level of probability to which the derived acceptance criteria would need to be met. For example, a 98th percentile is the target for neutron over power trip analysis; the USNRC employs a 95th percentile for large LOCAs in LWRs. The AECB Staff's general expectations are that

- the probability level / percentile will be consistent with levels established in previous applications and those used internationally (i.e. a 95th percentile or greater);
- some rationale will be established which, in some way, links the probability level to the frequency of the event; thus, for more frequent events (for example loss of flow), the analysis success criteria will be met with a higher level of probability;
- a high (but not necessarily quantified) level of confidence can be associated with the analysis.

In determining a level of confidence to which the analysis results meet an acceptance criterion, it must be recognised that a completely rigorous method is neither practical nor necessary: statistical rigour will not be achieved and all methods, at some level, rely upon engineering judgement and/or expert opinion. In addition, quantifying the uncertainty for some uncertainty contributors (for example, operator action in a small LOCA) can be difficult.

Statistical analysis of plant state. It is important that the key plant parameters be monitored and analysed so that trends in plant operation can be understood and can be taken into account in the analysis. There is an important distinction between stochastic variation in plant state and deliberate operation away from an "operating centre". For example: flux tilts and moderator poison levels can occur during startups that rarely occur during operation at power; the plant state during power manoeuvring; and operation in shim mode. Such plant states must somehow be covered in the safety analysis. Equally, it should be recognised that such occurrences are relatively rare and the probability of simultaneously being at the limit of the operating envelope is a condition that need not necessarily be analysed. At this time it is not clear what credit should be taken for infrequent but deliberate operation away from the operating centre. If any credit were to be taken, then operational procedures would need to exist that would return the plant state back to operating centre, within a specified time.

Compliance of plant operation to safety analysis assumptions. In the application of the operational parameter methodology to the LOF event analysis for Darlington, the key plant operating parameters were not set to limiting values. As a consequence, AECB Staff had several concerns regarding operational compliance for such parameters. The extent to which a best-estimate methodology will rely upon operational compliance is an important consideration.

A potential difficulty is the increased complexity of integrating the analysis with plant operation and ensuring that the analysis assumptions continue to be consistent with plant operating experience. Plant states would need to be monitored and the operating procedures would need to identify limits in duration of plant states and test intervals. To maintain the statistical validity of the analysis assumptions, plant parameter data would need to be recorded over a sufficiently long period time.
Phenomena and key parameter identification and ranking. Best-estimate methods require that important phenomena and parameters be identified and ranked. Ideally, a formal process should be used for the identification and ranking; the process needs to be scrutable and auditable. A formal process does not exclude the use of expert opinion or engineering judgement, however these should be used only when it is really necessary. In developing a process, recognition should be given to the interfaces between the individual technical disciplines (reactor physics, fuel thermal-hydraulics, etc) and the need for consistency. The ranking levels (for example, High, Medium and Low) need to be well defined. Also, the rationale for the treatment of each category (H, M or L) of parameter in the analysis needs consideration.

Determination of probability distribution functions (PDFs). Many best-estimate methods use PDFs to characterise the uncertainty in important phenomena and parameters. In deriving the PDFs we expect judgement / expert opinion to be minimised where possible. Where this is not possible, a scrutable and auditable process should be used. For plant parameters, where the PDFs are statistically derived from plant measurements, trends should be taken into account. For phenomena and models, the link between the PDFs and code validation needs to be made.

Extent of computer code analysis. Because of the substantial costs associated with computer code analysis, a surrogate for the computer code is frequently utilised in assessing the uncertainties. The surrogate may be a response surface or a much simplified fast running model. Since the surrogate is used to reduce the extent of computer code analysis, the surrogate must be validated against the computer code. The amount of code analysis required and how this validates the surrogate should be established in the methodology. We expect that more code analysis would be done at the high confidence level to support the surrogate in this region.

6.0 Final Conclusions

In conclusion, the trial applications have shown that realistic calculations and uncertainty analyses are useful alternatives/additions to more conservative methods. These applications have also highlighted a number of philosophical and technical issues that need to be considered in permitting best-estimate methods to become an established part of the licensing framework. Further work is required by AECB Staff to develop regulatory guidance, and by the licensees to fully develop their realistic analysis methods.

7.0 Acknowledgements

Thanks to Lorne Macdonald and Peter Elder whose frequent discussions on this and related topics helped to shape my thoughts and the content of this paper.

8.0 References

2. AECB Regulatory Document R-7, "Requirements for Containment Systems for CANDU Nuclear


STATUS OF THE FRENCH APPROACHES
FOR USING BEST ESTIMATE
CODES IN LICENSING

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ABSTRACT

In the safety cases which are analysed in the licensing process in France, there is a clear and increasing tendency to use Best Estimate (BE) Codes as part of the overall safety evaluation approach which includes the specific assumptions, the utilization of the code itself and the use of safety criteria. One example consists in the use of the thermalhydraulic code CATHARE but it can be seen that it should also concern several applications of codes in other domains such as severe accident. Application of BE thermalhydraulic codes in licensing have been made till now on specific questions (small break and intermediate break studies, emergency procedures). Examples will be given and the characteristics of these applications will be discussed. With the future reactor EPR, the BE approach will certainly be generalized and a more systematic approach for licensing will have to be defined. Consequently one can say that the use in France of BE methods in licensing is now in an intermediate state, but that it will certainly make progress and will go towards a more precise formalization in the next years.

In order to be prepared to this evolution, several research programmes are underway. A general study of uncertainties methods is performed in view of the definition of an "IPSN approach". Mathematical tools are developed which are applied to CATHARE but also to several other severe accident codes. A research programme has been initiated inside CATHARE project in order to evaluate the uncertainties of the elementary individual physical models. Research type applications are performed such as the UMS exercise in the framework of CSNI, in order to evaluate which problems are still to be solved and how they can be managed. All these research programmes are contributing to the extension of a more complete and extensive use of BE methods in licensing.

1. INTRODUCTION

The review on an international scale [1] of the use of Best Estimate (BE) thermalhydraulics code in safety analysis shows that there is already a very large number and a very large variety of applications which are performed using BE codes. BE codes are used in several safety cases such as LOCA, transients, RIA, containment and fuel behavior. These safety cases relate to reactors in the design phase as well as to operating reactors. Most analyses of present reactor events for example are now performed with BE codes. The same occurs for the development of accident management strategies including the verification of the effectiveness of backfit measures and the analysis of the emergency operating procedures. PRAs constitute another domain where there is an almost exclusive use of BE codes. Large number of calculations for core design, related to steady state core analysis and to subchannel analysis, are using since many years BE methods. Moreover BE codes are now going to interfere very often in nuclear power plant simulators and training.

In the licensing area, as shown by a recent study on the utilization of BE methodology in safety analysis and licensing inside CSNI countries [2], there are already some licensee calculations performed with BE codes. Most regulators permit now quite generally the use of BE codes in addition to the conservative appendix K approach. However if their requirements are converging when restricted to the general terms, one can observe that they deviate and become more and more qualitative when going further in the application. This demonstrates that in such applications there are still unsolved questions which should be clarified. Anyway the pressure to use BE codes is world-wide increasing, very often because reduced conservatism’s are required for
handling new plant situations or for developing new plant designs. In the case where the use of BE codes is not felt possible to be ultimately introduced in the licensing process, they can at least be considered as tools which can be used for auditing the licensee calculations, by verifying for example the conservatism's of Evaluation Models (EM).

The French approach for using BE codes in licensing and more generally in safety analysis follows well this broad international evolution: starting from a conservative approach based for LOCA on Appendix K rules, the use of BE codes is more and more progressing in the safety studies. The first application concerned non-DBA cases, such as for example the validation of Emergency Operating Procedures. Now, with new fuel management and with the design of EPR reactor, attempts are made to apply BE codes to DBA cases. The most recent applications of this kind are under evaluation and technical position are still to be defined for answering to some generic questions. In order to get the necessary knowledge for the introduction of BE codes in licensing, research programmes have been initiated, either on the methods for deriving the code uncertainties, or on the evaluation of those uncertainties in the framework of the French thermalhydraulic code CATHARE.

2. GENERAL USE OF SAFETY CODES

Before describing the French status of BE codes in safety analysis and before giving an outline of the research programmes in France, we will recall some of the general features related to the use of codes in safety analysis.

For several safety studies, the plant behavior during an accident is the key point as it determines the radiological consequences and as it conditions the safety measures to be taken. Most often it is not possible to predict this behavior only by experiments because experiments at scale 1 are not feasible and because one cannot extrapolate directly the experimental results from reduced scale to scale 1, due to an insufficient knowledge of the scaling laws. The use of codes turns then to be mandatory. The codes should include complete analytical models which, by describing the phenomena, will be able to make the scale extrapolation. In most cases, scaling and codes are consequently intimately linked, and this is the main explanation why the codes are so important in safety analysis.

If one starts very often the presentation of the use of codes by the conservative approach, one should not forget that the very first objective of every code developer is in fact to try to describe the physics in order to predict the phenomena occurring in the plant. In a first step this is clearly a BE approach and this approach has been used since decades. Of course, if BE codes are developed, it is in view that they will be applied to practical plant calculations. The most common way which has been and which is still being used, is by performing in a deterministic way, a unique code calculation which is claimed to provide the plant transient behavior. This mode of use has been defined in [3] as the Ordinary Direct (OD) use. In this OD use the calculation results are considered as "THE" response of the plant. They are directly used for drawing safety conclusions. A qualitative judgement on the code applicability to the calculated transient and some sensitivity studies are sometimes but not always added to this process.

In the 70's, when it was realized that the Best Estimate capabilities of describing the physical phenomena was too coarse and incomplete, a conservative approach was defined. The basic reason for this approach was to circumvent the lacks of knowledge of the physical phenomena. For this purpose two notions were introduced, the notion of consequences and the notion of criteria, majored for the first ones and restricted for the second ones. In order to encompass the code limitations, conservative assumptions were taken everywhere a phenomenon was insufficiently known. These assumptions could be either global majoration of consequences, or could consist in applying penalizing boundary conditions or penalizing assumptions in the models. By this way the accident consequences were maximized and as a result an upper safety bound which should not be exceeded was obtained and was then compared to the safety criteria.

The formalization of such an approach has been done by the USNRC in the early seventies, and has given the well known 10 CFR50.46 appendix K which has been largely and world-wide used in LOCA analysis.

It was clear that such an approach had several limitations (difficulty in demonstrating the conservatism's, impossibility to get an evaluation of safety margins, question of the additivity of the conservatism's...). Consequently simultaneously to the conservative approach definition, a broad research programme was launched internationally in order to elaborate what was called the Best Estimate codes, but which were in fact
new advanced BE codes (second generation codes) which will replace the less sophisticated codes (first generation) such as RELAP4 used also in Best Estimate ways. In fact these new BE codes were initially foreseen to gather all the best of the physical knowledge and consequently to be able to give realistic predictions. When those large thermal-hydraulic system codes like ATHLET, CATHARE, RELAP5 and TRAC have been made available, it has been realized that their predictions presented a lot of progress compared to the former codes but that nevertheless their realism had limitations since there remained necessarily some unknown in the physics. In order to use these codes by comparing their results to quantitative safety criteria, it was necessary to introduce the notion of uncertainties.

The idea is that the codes predictions present uncertainties due to the several sources of inaccuracies which have been introduced during the code development process. The evaluation of those uncertainties should give bands of values in which the real physical parameters should stand (with some degree of confidence when using statistical methods). In this Calculated Uncertainty (CU) use of the codes [3], the safety evaluation, such as for example the comparison with safety criteria or the definition of safety measures, is performed on the basis of those bands of values. The mean value, the upper or the lower band will be used depending on the safety question. The CSAU method developed by USNRC has been a precursor of this CU use of BE codes. Now several uncertainties methods (UMAE method, GRS method, IPSN method, UK method,...) are potential candidates for the CU use of the codes.

Finally there are today three main categories of use for safety codes [3] which are coexisting:
- the CO (conservative) use where the basic principle is to determine a conservative response by taking conservative assumptions
- the CU (Calculated Uncertainty) use where an evaluation of uncertainties give prediction of parameters accompanied with their error bands
- the OD (Ordinary Direct) use where a unique calculation is performed and where the response used corresponds in fact to the mean value obtained in the CU use.

3. STATUS AND TENDENCIES IN USING B.E. CODES FOR SAFETY EVALUATION IN FRANCE

3.1. Status

In France the mode of use of safety codes has in fact quite well followed the evolution which has been described just before.

The starting point has been the Conservative approach based, for LOCA, on the Appendix K rules. During several years in the 70s and in the beginning of 80s, the codes used for the safety demonstrations were licensed codes from the licensing vendor. Those codes were codes which received agreements from the USNRC. Consequently the safety evaluations for the safety authorities were focused essentially on the verification that the appendix K rules were correctly applied. This verification was performed on a case by case basis, and was including the verification of the boundary conditions, the review of the assumptions done in the calculation, the check of the effective agreement of the code used, and the verification of the adequacy of the results compared to the safety criteria.

This methodology based on a conservative approach following appendix K for LOCA analysis, has been used for the design of all the French plants which are now under operation. Within this method, the vendor Framatome introduced modifications when he decided to elaborate its own French system of codes (MEFRA) for large break LOCA analysis instead of the licensed codes agreed by NRC. On the basis of the preceding series of codes, modifications were made, especially improvements of specific models which consequently were less conservative than the old ones. For the safety evaluations performed in support to the safety authorities it was not possible anymore to remain only with the follow up of the appendix K rules, but advice on the model modifications were to be stated. Those modifications were based on the results of recent research programmes. In order to audit them, it was proceeded to comparisons with the BE code CATHARE and to the analysis of the experimental validation. The plant predictions with the modified code system was found acceptable but always as usual within the overall safety evaluation including the evaluation of the particular assumptions, the utilisation of the modified code itself and the use of safety criteria. This was the first interaction in France between conservative and BE approaches where the BE approach was used to verify the proposed reduction of conservatism.
Besides the design basis accident a lot of safety studies concerned the definition and the justification of the emergency operating procedures. These studies which became on the spot after TMI, require the prediction of the real plant behavior. Therefore Best Estimate codes have been used quite early in the OD manner. The codes were initially first generation codes (FRARELAP, AXEL, ....) until the advanced code CATHARE became available and could be used. Similar practice was used for transient studies but with specific transient codes (for example THEMIS). Again the OD use of the codes was applied for those transient predictions.

For all those last applications performed with BE codes in a non conservative way, most of the results were to be judged more or less qualitatively. The code validation which should show that the code was really suitable for predicting the phenomena at least qualitatively became very important and the safety evaluation should focus on its adequacy. The safety evaluation were performed on a case by case basis. The general overall safety evaluation included the review of the specific assumptions used for the calculation; the assessment of the calculation itself but for those particular cases, with a specific attention on the code validation, and the evaluation of the results with regards to the safety criteria.

At that stage we should mention the special case of the subchannel core analysis. These analysis have been performed almost always with BE codes such as the French code FLICA. These codes are used in an OD manner. However their main results which are the occurrence or not of the DNB (Departure from Nucleate Boiling) are treated statistically based on the experimental statistical dispersion of the calculation versus the experiment. This statistical treatment with the application of a safety criteria on the DNB (DNB Ratio) is quite similar to the uncertainty calculations and the use we would like to get. The only significant difference is that it has been applied tenths of years before the first applications of the CU use of the other thermal-hydraulic BE codes.

In France the first application of BE codes in DBA cases has been experienced in the GEMMES project. The objective of GEMMES is the increase of the length of the fuel cycle in the French 1300 MWe plants. This increase required to repeat some thermal-hydraulic safety studies with a new set of accident rules with regard to decoupling criteria to be fulfilled on the basis of the physical analysis of the transient. In particular the studies on small and intermediate breaks and on the long term phase of the LOCAs had to be repeated. For performing these safety analysis, the utility EDF has used exclusively the BE code CATHARE. For this purpose they followed a method that they developed and which is based on:
- first, the identification of dominant parameters,
- secondly, on the basis of validation on experiments, the determination of conservative values for these parameters,
- third, the calculation of the plant transient with all dominant parameters taken at their conservative extrema (one calculation).

The results of those GEMMES calculations have been evaluated by IPSN in support of the French Safety Authorities. The evaluation has been made on a case by case basis including the overall safety evaluation (assumptions, code calculation, criteria) as IPSN has always proceeded. The particular results which have been presented have been found acceptable, but no general agreement on the method itself has been given because it would need a quite long investigation which was not possible to do in the imposed time frame. A detailed presentation of those calculations and of the safety evaluation performed by IPSN will be given at this seminar and can be found in the paper [4].

### 3.2. Tendencies

The introduction of the use of a BE code (CATHARE) by the French utility and the French vendor, for safety analysis of DBA will undoubtedly continue in the future. Today similar studies as GEMMES are underway in EDF and Framatome with the CYCLADES project. Consequently, it is quite certain that old appendix K approach will be less and less applied.

For the project of European Pressurized Reactor (EPR), it has been clearly recommended by the French and German safety authorities that: "realistic assumptions and models could be used" ....... "but the compliance of the results with the acceptance criteria must be proven at a high confidence level - this implies the use of a frozen version of the computer code which has to be qualified and verified and an explicit evaluation of the associated
uncertainties combining the elementary uncertainties (code models, scaling effects, initial and boundary conditions, user effects,...). An alternative approach could be the use of models and criteria already applied to existing plants in a conservative way."

Presently, the EPR project (NPJ) is developing all its safety calculations following a Best Estimate approach. Two papers in this seminar [5], [6] will illustrate the attempts done for using BE codes in the design of EPR.

These examples show that an irreversible change is taking place in direction of the use of BE codes in safety calculations. Presently different methods to handle uncertainties are tested and used. Till now we have not reached application of a fully Best Estimate method with a complete uncertainty analysis and evaluation. The method used in GEMMES, for example can only be considered as an interim one as it uses a BE code (CATHARE) for determining dominant parameters but it applies then conservatism to those dominant parameters instead of doing a complete uncertainty study.

Of course areas where BE codes were already in practice will continue to be examined in a BE way. This is the case for the EOPs studies, for some operational transients prediction, for beyond design basis accident (BDBA), for the PSAs calculation, etc.....The only modification we could expect is that they will include sometimes evaluation of uncertainties. These evaluations will be added, in a way to be defined, to the present physical predictions which are now generally obtained by an OD use of the code. This will help surely in answering apparent inconsistencies which appear quite often in the utilisation of the results.

In conclusion, the tendency in licensing in France is clearly oriented towards the use of Best Estimate methods. Till now, the approach has not gone the full way and we can consider that we are in an intermediate state. The objectives and the challenges will certainly push in the direction of more Best Estimate practices. On the safety evaluation side this will force to a more precise formalization of the requirements to be fulfilled for those best estimate methodologies to be acceptable.

4. RESEARCH PROGRAMMES IN VIEW OF USING B.E. METHODS IN SAFETY ANALYSIS

In order to be prepared to the evolution towards Best Estimate methods in licensing, IPSN has developed research programmes on the use of Best Estimate methods and particularly in view of evaluating code uncertainties. This research programme has been developed in six directions:
- A general analysis on the method to be used for deriving uncertainties of the plant calculation.
- The determination of a method for evaluating the response uncertainties
- Mathematical tools for performing easy statistical treatment of all data.
- The programming inside CATHARE of the Discrete Adjoint Sensitivity method (DASM)
- An evaluation of the uncertainties of the elementary individual physical models
- A first global application of the method by the participation to the UMS exercise.

4.1. General IPSN method for the evaluation of uncertainties

This method has been derived in order to get a general view on the different steps to be performed for obtaining ultimately the uncertainties of the plant predictions. The flow chart (see table 1) which has been defined constitutes a framework more than a definitive method. However, the principles are well set up, even if some details may change in connection with the application feedback. This method has large similarities with the GRS method [7] as it was developed in collaboration. Differences however subsist in several details which should be validated by practical applications.

In order to explain the flow chart of table 1, we may recall shortly the problematic of the uncertainties evaluation (see [1], [3]). First we should go back to the formulation of the physical model.

If X is the vector of the state parameters of the physical system, the equations of the model can be written:
(1) \( F(X, C(X)) = 0 \)
where the transfer laws C can be written as function of the vector X and of empirical coefficients \( e_1 \):
(2) \( C = C(X, e_1) \)
If we add the boundary conditions:
The solution of the model can be written as
\[ X = X(e, l_i, z, t) \]

The modelling process consists in establishing first the F and the C functions. The correlation work which follows consists in adjusting the coefficients \( e_i \) on experimental data namely on separate effect tests (SETs). Qualification will give a judgement on the quality of the \( e_i \) using the SETs, whereas verification on Integral tests (ITs) will provide a judgement on the global quality of the solution \( X = X(e, l_i, z, t) \).

The analysis of the equations (1) to (4) in terms of physical uncertainties, gives logically the following sources of code uncertainties:

- The first source is the uncertainty in the formulation of the functions F and C. This uncertainty is mainly due to the simplifying assumptions which have always to be made during the derivation of these functions.
- The second source consists of the uncertainties in the determination of the coefficients \( e_i \) and on the knowledge of the boundary conditions \( l_i \).
- The third source comes from the uncertainties in the resolution of the equations which are related to the numerical methods.
- The fourth source is made from the errors related to the way the code is used (user effect).

As a consequence of these uncertainties sources, three steps can be distinguished in the process of uncertainty evaluation:

- Step 1 where one verifies the "adequacy" of the formulation of \( F = 0 \) and of C
- Step 2 where one evaluates the uncertainties in the adjustment of \( e_i \) and in the determination of \( l_i \)
- Step 3 where, from the \( \Delta e_i \) and the \( \Delta l_i \) one evaluates the uncertainty on the response \( R(X, e_i, l_i) \).

We can find these 3 steps in the flowchart of the method given on table 1 where they are numbered I, II, III.

**Step I**

This step (see table 1) is a qualitative one which should result in a statement about the code applicability. This applicability is related to a defined frozen version of a code, to a type of plant, and to categories of scenarios for which the code has been developed. On one side one should identify the physical phenomena to be described, on the other side one should identify the different models. The intercomparison of models and phenomena will determine the code applicability depending on the quality with which the models are describing the phenomena. This determination will be based on the analysis of the code validation results and on the completeness of the validation matrix which should cover the different plant phenomena to be described. In the flow chart the code applicability is determined scenario by scenario. This does not mean that the applicability can be determined for only one scenario. It is clear that in the judgement of the model adequacy, the capabilities of handling different scenarios is crucial, otherwise we can suspect that the models are in fact well tuned for the given scenario. This step is very important as it conditions the chances of success of the following steps. If the applicability is not sufficient, it means that the equations of the models F and the physical laws C are insufficient and have to be revised. In this case all the analyses which are following and which are based on the use of F and C will be pure nonsense.

**Step II**

The main body of this step is represented by the central line on table 1. The first action is to identify the parameters \( e_i \) and to rank them. For practical reasons it will not be possible to perform the following process for all parameters, even if the chosen methods allow the use of very large numbers without any additional cost. The ranking should allow to eliminate the parameters which contribution is really negligible. The second action will be to determine the uncertainties \( \Delta e_i \). For this, the separate effect tests will be used as they are normally designed for giving all necessary information on the specific model. In the next action, the parameter uncertainties are propagated with the model F, using statistical assumptions, for the prediction of system tests. Comparisons with the experimental data provide a global verification of the approach and allow the final determination of the uncertainties related to the code parameters.
Besides the central line we have two lines which treat uncertainties which cannot be handled in the standard way. On the left side (table 1) we have all the uncertainties due to the model deficiencies. These deficiencies can come from the fact that some phenomena are not modelled (this will always occur even with the best code) or from the fact that in the determination of the parameter uncertainties it will appear some model insufficiently qualified. For those deficiencies a special evaluation will be necessary in order to determine their contribution to the overall uncertainty. On the right side we have in a similar way the uncertainties related to scaling. They are analysed using SETs and ITs as far as possible contributing by this way to the specific evaluation of their contribution.

Step III

In this final step, the uncertainty on the response R is obtained by plant calculations which provide the propagation of the individual uncertainties to the response R. These individual uncertainties include (see table 1) the model parameters uncertainties $\Delta \eta$, the plant uncertainties (mainly the boundary conditions $\Delta l_i$), the uncertainties related to scaling, and other sources of uncertainties (numerics, user effects, ....).
4.2. Development of a method for evaluating the response uncertainties

In the principle of the IPSN method, the uncertainty of the response \( R(X, z, l) \) where \( X \) is the solution of the system (1) \( F(X, C(X)) = 0 \), is obtained by the propagation of the elementary uncertainties \( \Delta z \) and \( \Delta l \) through the system (1). For that purpose, the elementary uncertain parameters \( z \) and \( l \) are considered as they were random variables inside their uncertainty bands. For each of them one defines a range of variation and a subjective probability density function (SPDF) which express the range of values where the uncertain parameter may stay. Hence the response \( R(X, z, l) \) where \( X \) has the mathematical form (4) \( X = X(z, l, z, l) \), can also be considered as a random variable. The uncertainties propagation problem being transformed in a statistical problem, one can determine the uncertainty ranges by theorems of statistics and one can evaluate the PDF of the response by usual statistics analysis techniques. In the principle, one takes samples of the set of uncertain parameters \( z \) and \( l \). For each of these samples the code calculation provides a solution \( X \) and a value for the response \( R \). Repeating this operation for \( N \) samples, one obtains the distributions for \( X \) and \( R \) which can be afterwards treated statistically. In order to reduce the number \( N \) of samples one uses a mathematical theorem called the Wilks formula which calculates the minimum number of samples necessary to obtain the uncertainty band of \( R(X, z, l) \) with a given probability and this with a given confidence level. The interest of this method is that the number of samples does not depend on the number of parameters. For example to get an uncertainty band where 95% of values of the response are included with a 95% of confidence level, the number of samples may be at least 93 whatever the number of uncertain parameters is. This method has been originally proposed by GRS [7].

4.3. Development of mathematical tools: SUNSET

To manage all the statistical treatments, a mathematical tool has been developed called SUNSET [8]. In a first step, SUNSET is handling the sample generation:
- it manages the uncertainties of the parameters as input. The SPDFs can be defined using different standard laws such as normal, lognormal, uniform distributions, ...
- it generates the samples with the choice of using different statistical techniques, such as simple random sampling, Latin hypercube sampling, ...
- it can take into account the dependencies between the uncertain parameters which are not independent.

In the second step, SUNSET manages all the calculations which will generate from the samples defined before, the distribution of the responses which the user has requested.

In the third step, SUNSET provides several capabilities for the statistical treatment of the preceding responses.
- It handles all the usual treatments like calculation of extrema, visualisation of individual or all responses, comparisons with experiments, ...
- it performs evaluation of PDFs of the response by using fit tests to known probability laws (two methods available: \( \chi^2 \) tests and Kolmogorov-Smirnov tests) or by optimisation techniques (Pearson, ...)  
- it calculates response sensitivity measures to the parameters by using different statistical coefficients: standardized regression coefficient, partial linear correlation coefficient, partial rank correlation coefficient. The responses which can be treated by SUNSET can be scalar responses or time dependent responses.

The software SUNSET can be coupled with every kind of codes. Besides its first objective of uncertainties propagation, it is a very powerful tool for performing sensitivity studies particularly for ranking models and the corresponding phenomena. SUNSET has been coupled to CATHARE and to the severe accident codes, ICARE, ESCADRE, CIGALON, MC3D.

4.4. Development of the Discrete Adjoint Sensitivity Method inside the CATHARE code

In the different processes of uncertainty evaluation, it could be interesting to have the sensitivity of the response \( R \) to the uncertain parameters \( z \). One way to get the derivative \( \delta R / \delta z \) is to run two codes calculations by varying the parameter \( z \), of + or - \( \Delta z \). The variation of the calculated response \( R \) gives then the derivative which is wanted. This shall be repeated for as many as \( z \) considered. Another way is to use the so called adjoint sensitivity method. [9], [10]. This method has been implemented on the discretized system. If \( F^d \) is the
discretised form of the function $F$ of the model (1), one solves the following linear adjoint system for each time step:

$$
\begin{align*}
\frac{\delta F^n}{\delta X^n} \cdot \Phi^n &= -\frac{1}{\delta} \frac{\delta F^{n+1}}{\delta X^n} \cdot \Phi^{n+1} + \frac{\delta R}{\delta X^n} \\
\end{align*}
$$

provided that during the direct calculation one has stored on file the values of $X^n$ and of $\delta F^n / \delta X^n$.

The result is given by:

$$
\begin{align*}
\frac{d R}{d e_i} &= \frac{\delta R}{\delta e_i} + \sum \Phi_n \cdot \frac{\delta F_n}{\delta e_i} \\
\end{align*}
$$

This method has been implemented in the standard version of CATHARE 2 V1.4. This means that all the necessary derivatives are stored during the direct calculation. The evaluation of sensitivities is then obtained by a post processing of the CATHARE 2 files. This method is very inexpensive in CPU time: in one code calculation one can obtain just by solving linear equations as many response sensitivities as required.

### 4.5. Evaluation of the uncertainties of the elementary individual physical models

From the flow chart of the IPSN method, one can easily realize that the whole uncertainty evaluation process is strongly dependent on the determination of the uncertainties of the elementary parameters of the individual models. The values of these elementary uncertainties are conditioning in fact entirely the values of the global response uncertainties.

The evaluation of the physical models uncertainties shall be done by comparison with experimental results. The best experiments for this evaluation will be the separate effect tests which have been designed for the development of the corresponding model and their qualification. The uncertainty evaluation of the elementary physical models should have then a very strong connection with the code qualification process. It is for this reason that inside the CATHARE development an evaluation of model uncertainties has been initiated in parallel to the qualification of the next version of the code (CATHARE 2 V1.5 rev6). In order to reach this objective, a methodology (CIRCE) for determining the uncertainties of the constitutive relationships has been developed [11], [12]. Assuming that the uncertain parameters $e_i$ are random variables which follow, via an exponential multiplier, normal distribution laws, the methodology is evaluating the covariance matrix of the random variables defined before. For this evaluation, a statistical analysis of the differences between the true response, the experimental response, and the code response is being made. The experimental response is taken from the separate effect tests for which the response is sensitive to the physical models which are analysed. Using the sensitivities of the code response to the uncertain parameters obtained by the Discrete Adjoint Sensitivity Method (DASM), using also the difference between the code and the experimental response, and using finally the Bayes' theorem, the covariance matrix is calculated iteratively. From the covariance matrix one can calculate the standard deviation of the uncertain parameters and the coefficients of correlation between them.

This method is now being applied on CATHARE 2 V1.4E in parallel to the systematic qualification of the code.

Other alternative methods for evaluating the elementary model uncertainties have been applied or are in preparation at IPSN. In order to get a complete preliminary set of values of the uncertainties of the parameters $e_i$ for performing the UMS exercise, IPSN has reviewed the main separate effect tests in the CATHARE validation matrix. On these tests, simple sensitivity studies have been performed and the uncertainty ranges have been empirically evaluated by bounding the experimental values. The uncertainty bands which have been obtained must be considered as preliminary, but they allow already to perform calculations and can be used as starting basis for further improvements. An other attempt is being made for developing a method which allows direct evaluation and check of the elementary uncertainty ranges and which will be based on the use of SUNSET on adequate series of separate effect tests.

In conclusion, a significant effort is being done in France on the evaluation of the uncertainties of the elementary individual physical models. We are in the middle of the process but it is clear that the final results
will be crucial for the credibility of the whole uncertainty evaluation approach and consequently on the real possibility to apply Best Estimate Methods in the licensing.

4.6. Example of application: the French contribution to UMS

The first complete application of the IPSN method has been made at the occasion of the UMS exercise (Uncertainty Method Study) organised by CSNI/PWG2. The transient on which the uncertainty evaluation has been made was a ROSA IV-LSTF test provided by JAERI. The test (SB-CL-18) simulates a small break LOCA (5%) without high pressure injection. Two core uncoveries were experienced during the test, one due to manometric effects, the second due to the lack of mass in the primary circuit.

The results have been extensively published [13], [14]. From the uncertainty ranges of the elementary models established as explained above, the SUNSET software has been applied. It gave the uncertainty bands of all the required response without any limitation in the number of uncertain parameters taken into account. It gave also sensitivity measures from which without any subjective engineering judgement, the models which are the most important for the given transient can be determined.

Finally this exercise has shown the applicability of the uncertainties propagation method. On the general flowchart of the IPSN method, it demonstrates that the central boxes of step II i.e. identification and ranking of parameters, quantification of parameters uncertainties (separate effect tests), verification and coupling of model uncertainties (system tests) are feasible actions. As counterpart this enhances the boxes which have not been checked and which represent open questions still to be solved, like the evaluation of the uncertainties due to the deficiencies of the code and the evaluation of the uncertainties related to the scaling.

5. CONCLUSIONS

To summarize the status in France about the use of Best Estimate codes in licensing cases, we can say that there is a more and more pronounced tendency to go in this direction in the licensing process. Vendors and utility are developing several methodologies for applying BE codes to their licensing cases. Only one case for DBA has already been treated in a licensing procedure with a BE code. But as it is based on the use of conservatisms, it does not constitute consequently a fully best estimate approach. Many other cases using BE codes are in preparation.

At the same time there are several areas which were and which will continue to be treated by BE codes in the Ordinary Direct Use manner (for example emergency operating procedures, severe accidents, PSA,...). These areas have not integrated the notion of uncertainties. In many cases qualitative answers are enough and the present approach is quite satisfying. But when questions are raised on the quantitative results, one can observe very often conflicts which in fact are mainly due to the absence of uncertainties evaluation.

In conclusion on the licensing side, we are in an intermediate state, beginning to leave the appendix K conservative approach and not yet in a fully Best Estimate approach.

On the research side, efforts are made in order to be prepared to the introduction of the complete Best Estimate approach in licensing. In IPSN, an analysis has been made in order to define a logical and rigorous approach of uncertainties evaluation. This analysis has produce a flowchart (IPSN method) which gives a general framework. This method is based on the propagation of the uncertainties of the elementary parameters of the model. An efficient mathematical tool (SUNSET) has been developed in order to handle this propagation and to perform the related statistical-treatment. Several investigations on the determination of individual elementary uncertainties are underway. They give encouraging results but the studies should be continued quite substantially.

Looking to the future, it appears that on the uncertainties evaluation there are still open questions such as scaling, non modelled phenomena, user effect,.... For those sources of uncertainties, quantification will have to be supplied in the future if we want to finalize the process. When the uncertainty question will be clarified, the application of the BE codes in the licensing process itself, with the full uncertainties evaluation, will raise additional questions before that we could consider to have reached a fully Best Estimate licensing method.
Using uncertainty bands will require for example a new formalisation of the fulfilment of the safety criteria. As a second example, the licensing process will become much more complex and will use quite sophisticated methods. Consequently it will certainly raise several questions about the control of all those procedures and this should have significant consequences on the general QA of safety analysis. Nevertheless when fully Best Estimate methods will be really applied to licensing, it will constitute a very important progress in the use of computer codes in the safety analysis and a large valorization of more than twenty years of research.

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Acknowledgement: The authors would like to thank Mr. E. Chojnacki and Mr. A. Amri for their kind review of the paper.

79
ABSTRACT

The applicant in the German licensing process supplies the deterministic thermal-hydraulic safety analyses to the state safety authority. Independent analyses are conducted for confirmation by the safety experts as part of the safety assessment. Besides the safety principles laid down in the legally binding compilation of rules released by the Federal Ministry in charge of the safety of nuclear installations, the detailed guidelines of the reactor safety commission concerning the assumptions for the calculations provided the latitude necessary for the consideration of the further development of safety technology. The applicant, also the safety assessor, can demonstrate that other alternative assumptions, models or correlations will assure safety in at least an equivalent way. These demonstrations are usually based on reliable experimental results. Hence procedures were used in the Germany licensing practice which contained conservative and best estimate features from the beginning. A pure evaluation methodology with licensed assumptions and frozen codes was not a specific requirement.

After extension of the experimental data base and after considering them in the further development of code models, the earlier conservative calculations were replaced by more realistic calculations. At the occasion of reassessment of safety of existing plants new best estimate calculations were performed. The comparison of the new results with the earlier results identified great differences. At present the thermal-hydraulic system codes are best estimate codes based on comprehensive validation including
experimental results obtained in differently scaled test facilities up to large scale facilities. Conservative initial and boundary conditions are used for licensing cases.

In parallel to the trend towards more realistic best estimate calculations, methods for the quantification of uncertainties of the calculated results were developed and tested. Binding regulations for the uncertainty quantification do not exist yet. The use of realistic assumptions and models for the safety demonstration of future PWRs was commonly recommended by the French and German advisory bodies GPR and RSK to the Ministries in charge of reactor safety in the two countries. These recommendations were adopted by the French German Directorate DFD (Deutsch Französisches Direktorat) the common body of the ministries. They recommend it under the condition that the compliance of the results with the acceptance criteria is proven at a high confidence level which means also the explicit evaluation of the associated uncertainties. DFD would also allow the use of models and criteria in the conservative approach already applied to existing plants. This alternative, however, is not preferable because the high level of code validation could not be utilized in this case. In future, therefore, the safety evaluations will preferably be based on full best estimate methodology. The prerequisites for this methodology consist of a high standard of code validation, qualified users and a reliable quantification of uncertainties. This implies a realistic accident simulation with models reflecting the best available knowledge. Some of the initial and boundary conditions, however, will still follow the conservative concept as before, like assumptions of single failures and preventive maintenance of components.

Two methods for the quantification of the uncertainties are available at present in Germany. The designer's method follows essentially the CSAU approach, but differs in the application of some steps. It will be presented in this seminar by F. Depisch et al. from Siemens. The GRS method will be applied for the confirmatory analyses conducted as part of the licensing process by the expert organisations. This statistical method will also be presented in this seminar by H. Glaeser. The designer already uses his method. GRS completed the development of the method and the application on experiments and a reactor analysis. Work is going on to improve the knowledge on highly sensitive input parameters. At present GRS is prepared to apply their method using conservative uncertain input parameter distributions for those variables for which the data base is not yet validated sufficiently.
1 Previous Practice in Licensing

At present 20 commercial Nuclear Power Plants (NPP) exist in Germany, six plants with Boiling Water Reactors (BWR) and 14 plants with Pressurised Water Reactors (PWR) out of which one plant (KMK) is not allowed to operate due to a legal problem. The PWRs can be divided into four generations, the plants of the fourth generation are the plants of KONVOI type. The two types of BWRs are the 69 line and the 72 line. The contribution of nuclear power to the total electricity production in Germany was 35% in 1997, within the so-called base load range (electricity supply around the clock) even 60%. In six of the federal Länder, the share of nuclear energy in the electricity generation for the public supply grid was more than 50%.

1.1 Overview on the Licensing Procedure

A public utility, in most cases in co-operation with the plant vendor, provides his application to the licensing authority of the Land where the plant is to be sited for a license to construct and to operate a NPP (Fig. 1). The documents which have to be attached to the written application are specified in the Nuclear Licensing Procedures Ordinance (AtVIV), see Fig. 2. An important document is the safety report. The facility and its operation as well as the connected effects, including the consequences of design basis accidents, are explained and the countermeasures are described in this report. The safety analyses as part of this report were documented in the application for the KONVOI plants in three handbooks: the ECC Handbook (loss of coolant accidents = LOCA), the Plant Dynamics Handbook (Transients incl. ATWS), and the Core Design Handbook (reactivity initiated accidents = RIA). In examining the application the authority is advised by experts, in general by a Technical Inspection Agency (TÜV). At the same time the supreme federal atomic authority, until 1986 the Federal Ministry of the Interior (BMI), from 1986 the Federal Ministry for the Environment, Nature Conservation and Nuclear Safety (BMU), its responsible authorities, as well as central, regional, and local authorities concerned are involved in the process.

It is one of the important task of the licensing authority to involve the general public in order to protect the basic rights of those citizens who might be affected by the effects of the planned NPP. An important prerequisite for the legality of the decision of the authority is the compliance with the procedural requirements
set in Section 5 AtVfV. After the decision has been made objectors can bring action before the administrative court. The licensing authority considers in the assessment the opinions and recommendations of the experts entrusted, the BMI/BMU comment, the comment of the participating authorities, the objections brought forward by the general public, and the results of the environmental impact assessment. The authority can reject the application or grant the license or grant a partial license. Four partial licensing steps were granted for the NPPs of type KONVOI:

- location, safety concept, construction of the essential buildings,
- construction of safety-related systems and components,
- handling and storage of fuel elements (FE), loading the reactor with FEs,
- final construction, first nuclear start-up and operation of the NPP.

The BMU/BMI is advised by the Reactor Safety Commission (RSK) and the Commission on Radiological Protection (SSK), which are composed of independent experts of different scientific disciplines. After examination, RSK and SSK give a recommendation to BMI/BMU which is thereafter analysed by BMI/BMU. BMI/BMU gives comments to the responsible licensing authority. These comments have to be considered in the decision-making process. In addition, Gesellschaft für Anlagen- und Reaktorsicherheit (GRS) mbH, being the main adviser to the government, provides expert opinion to BMI/BMU.

Because of the large scope of the examinations, generally expert organisations, like TÜV and GRS and further competent experts are entrusted with the assessment and the examination of the application documents and of the planned site for the NPP. As far as the safety analysis is concerned, the assessors critically reviewed the documents, especially the calculations in the three handbooks.

The main task of this first step was to check the fulfillment of the safety-related rules and guidelines as far as the selection of initial and boundary conditions as well as the model assumptions are concerned. Also the adequacy of the applied code system and the completeness of the spectrum of calculations to prove the effectiveness of the safety system is checked in this first step.
In the **second step** the assessors may perform their own calculations. The assessors calculations partly served to check the calculations of the applicant and partly to complete them. It is not a prerequisite in the rules and guidelines that applicant and assessor use different codes for their respective analyses, but it is practice in Germany that in most cases the assessors were applying different codes than the applicant. A higher degree of independence was reached thereby even though the codes had already reached a high and a comparable level. The assessor also checked the pant data base before using it for building up their input decks for the codes. The codes used by applicant and assessors in the thermal hydraulics safety analysis for the KONVOI plants are listed in the table below. The table also contains the codes which would be used if there would be a licensing procedure nowadays in Germany. These are the codes which are applied at present for the safety examination of operating NPPs in Germany.

<table>
<thead>
<tr>
<th>Type of Incident</th>
<th>Applicant KONVOI</th>
<th>Present</th>
<th>Assessor KONVOI</th>
<th>Present</th>
</tr>
</thead>
<tbody>
<tr>
<td>LBLOCA</td>
<td>LECK4/MOD2</td>
<td>SRELAP</td>
<td>DRUFAN</td>
<td>ATHLET</td>
</tr>
<tr>
<td></td>
<td>HYDRANS</td>
<td>COCO</td>
<td>FLUT</td>
<td>TESPA</td>
</tr>
<tr>
<td></td>
<td>REWAS</td>
<td>RODEX</td>
<td>TESPA</td>
<td>RALOC</td>
</tr>
<tr>
<td></td>
<td>WAK-3</td>
<td></td>
<td>TRAC-PF1</td>
<td></td>
</tr>
<tr>
<td></td>
<td>BETHY-AZ</td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>ISBLOCA</td>
<td>RELAP5/1</td>
<td>as for</td>
<td>DRUFAN</td>
<td>ATHLET</td>
</tr>
<tr>
<td></td>
<td></td>
<td>LBLOCA</td>
<td>RELAP5/1</td>
<td></td>
</tr>
<tr>
<td>Transients</td>
<td>LOOP7</td>
<td>SRELAP</td>
<td>ALMOD 3.5</td>
<td>ATHLET</td>
</tr>
<tr>
<td></td>
<td>NLOOP</td>
<td>+ EUMOD</td>
<td></td>
<td></td>
</tr>
<tr>
<td>Static nuclear</td>
<td>SAV79A</td>
<td>CASMO</td>
<td></td>
<td></td>
</tr>
<tr>
<td>design</td>
<td>PANBOX</td>
<td>QUABOX/CUBBOX-HYCA</td>
<td></td>
<td></td>
</tr>
<tr>
<td>Hydraulic design</td>
<td>THERMOHYDRAULIK</td>
<td>COBRA</td>
<td></td>
<td></td>
</tr>
<tr>
<td></td>
<td>COBRA</td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>Core transients</td>
<td>PANBOX</td>
<td>SRELAP + EUMOD + PANBOX</td>
<td>QUABOX/CUBBOX-OX-HYCA</td>
<td>ATHLET + QUABOX/CUBBOX</td>
</tr>
</tbody>
</table>

**Codes Applied by Applicant and Assessors**

*in Previous Licensing of KONVOI and in Case of Present Licensing*

In the **third step** the assessors evaluate the effectiveness of the safety system, e.g. the effectiveness of the ECCS. This evaluation is based on the first and second step. Also comparison with relevant experimental results play an important role. Depending on the results of the evaluation the assessors may recommend to the authority to re-
quest the fulfilment of additional requirements. Due to different codes, different models in the code, different nodalisation and partly due to different boundary conditions the results of the independent calculations of the assessors differed to a certain extent from the results of the applicant. As long as the results were close enough to confirm a sufficient safety margin with reference to the technical criteria no subsequent require-
ments were recommended. This was the case in the majority of evaluations.

An overview on the analyses which were performed by the applicants and by the assessors in the frame of the licensing procedure for the KONVOI plants is given the appendices 1 and 2. It is obvious that a large amount of independent check analyses were performed by the assessors. Besides the analyses performed under the assumptions prescribed in the rules and guidelines (see chapter 1.2 of this report), the applicant also performed an analysis for the design basis accident LBLOCA with 2A break size in the cold leg under "best estimate conditions" (see 3.3.1 in appendix 1). The quotation marks shall indicate that the differences between the "best estimate" and the "conservative" analysis is only in the initial and boundary conditions but not in code options for different models in the code or in model variables. The main differences between the two types of LOCA analyses are presented in the following table:

<table>
<thead>
<tr>
<th>Break opening time</th>
<th>&quot;best estimate&quot; analysis</th>
<th>&quot;conservative&quot; analysis</th>
</tr>
</thead>
<tbody>
<tr>
<td>Reactor power</td>
<td>100 %</td>
<td>106 %</td>
</tr>
<tr>
<td>Decay heat</td>
<td>DIN simplified</td>
<td>DIN simplified + 2 σ</td>
</tr>
<tr>
<td>Power shape (16x16)</td>
<td>3rd cycle, mid</td>
<td>1st cycle, 4 d</td>
</tr>
<tr>
<td>Hot spot factor</td>
<td>2.57</td>
<td></td>
</tr>
<tr>
<td>Gap HT coefficient- average rod</td>
<td>axially varying</td>
<td>axially uniform</td>
</tr>
<tr>
<td>- hot rod</td>
<td>depending on local power</td>
<td>5500 W/m²/K</td>
</tr>
<tr>
<td>Containment pressure</td>
<td>about 17000 W/m²/K</td>
<td>7500 W/m²/K</td>
</tr>
<tr>
<td></td>
<td>considering pressure reducing effects</td>
<td></td>
</tr>
</tbody>
</table>

Essential Conditions for LOCA Calculations (Licensing Procedure of KONVOI)
1.2 Influence of Rules on Safety Analysis

Most of the nuclear licensing procedures for German NPPs were conducted in the decade between 1976 and 1986 (Fig. 3). The latest NPP, GKN 2 (Neckarwestheim, block 2), went into commercial operation in April 1989. The safety requirements to be met by the NPPs were laid down in national laws, ordinances and further regulations and guidelines or directives of the BMI, often on the basis of recommendations of advisory bodies, and by technical standards.

The Atomic Energy Act (Atg) provides the legal prerequisites for the peaceful utilisation of nuclear energy. It was enacted in 1960. In the meantime it has been amended several times due to political and technical/scientific developments. Its major purpose remained unchanged, it is to protect life, health and property of the population against the dangers of nuclear energy as first priority over economic interests. If a license pursuant to Section 7 AtG for a NPP is to be granted, the licensing requirements pursuant to Section 7, paragraph 2 AtG are to be fulfilled in particular. A license of a NPP may be granted only if these requirements are fulfilled, among them the condition that “that the necessary precautions have been taken in the light of the state of the art to prevent damage resulting from the construction and operation of the facility”. In July 1994 these were amended substantially by the new subparagraph 2a which requires that NPPs have to ensure that even extremely unlikely events involving core melt-down would not require radical actions to ensure protection against the damaging effects of ionising radiation outside the enclosed installation site (fence). This licensing precondition does not apply to NPPs for which a license or a partial license was issued before 1st of January 1994. According to the German Constitution, the Länder are responsible for the implementation of the AtG. In order to ensure uniform implementation the BMI/BMU can issue directives concerning the legality and expediency.

The provisions of the AtG are supplemented or specified by further Federal Acts and Ordinances which are legally binding as well. There the safety requirements are not specified in great detail, allowing room for different technical solutions, which all have to meet the same protection goals. The most important ordinance for the safety analysis is the Radiological Protection Ordinance (StrlSchV). In sections 28, 44 to 46 and 48 StrlSchV the most important radiation protection principles are comprised. In section 45 StrlSchV the radiological limits in any point in the facilities surrounding by releases into water, atmosphere and population are defined for normal operation. In sec-
tion 28, paragraph 3 StrlSchV the radiological limits during possible accidents are defined. In the nuclear licensing procedure the main emphasis of the examinations is to prove that the radiation exposure even below these limits is kept as low as possible (ALARA principle).

There is a variety of safety regulations below the level of laws and ordinances which are to be considered within the framework of the nuclear licensing procedure. They serve the purpose of demonstration adequate provisions against damages. They include the BMI NPP Safety Criteria approved by the Länder Committee for Nuclear Energy and hence they are legally binding as well. The Safety Criteria contain concrete technical provisions to render the safety requirements for the design of the engineered safeguards to be provided to cope with incidents: redundancy, diversity, general avoidance of interlaced partial systems, spatial separation, fail safe operation of systems, and preference to passive over active engineered safeguards. To avoid misunderstanding because of the intended shortness of the Safety Criteria in particular with regard to the single failure concept, additional interpretations were officially issued in 1984. They include examples for the for the application of the Safety Criteria in accident analysis.

In 1982, the Incident Guidelines were issued. They apply to NPPs with PWR except for those for which the first partial construction permit was granted prior to July, 1982. Hence they were in force for the PWR of the fourth generation with type KONVOI. These guidelines determine the incidents on which the design of NPPs has to be based in accordance with section 28 StrlSchV, paragraph 3, the design basis accidents (DBA). In a first group the incidents with relevance of their radiological impact on the environment are concerned. The second group lists the incidents against which precautions in terms of engineered safeguards have to be taken and which because of the precautions that have been taken, are not relevant as far as their radiological impacts on the environment are concerned.

Besides the safety standards of the Nuclear Standards Committee (KTA) which comprises sample solutions and representing resources for the decision process of the authorities, the most important one among the safety standards for the safety analysis is the compilation of safety-related guidelines of the RSK (RSK guidelines). The legal status of these safety standards, as the status of other technical standards, like standards of Deutsches Institut für Normung (DIN), and further recommendations, in a par-
ticular licensing procedure is stipulated by the concerned licensing authority of the Land. The scope of application of these guidelines are examined on a case by case basis. Since their application was generally requested in the German licensing practice, the RSK guidelines have obtained a great practical importance. The guidelines allow principally alternatives if the applicant or the safety assessor demonstrates that other measures will assure safety in at least an equivalent way. This provision allows for the necessary latitude in the development of safety technology. Hence deviations from the guidelines were determined on a case by case basis. At the time of the licensing procedure of the KONVOI plants the third edition of the RSK Guidelines for PWR from October 1981 was valid. At present the same edition in its last version from October 1996 is valid which contains several modifications, mainly as far as I&C of the safety system is concerned. They have been published in the Federal Gazette.

The RSK guidelines contain, among others, the specification of technical criteria of the acceptance of safety analysis, i.e. the decoupling criteria for the thermal hydraulics system analysis to be observed in safety analysis. For the analysis of loss of coolant accidents (LOCA) for the demonstration of sufficient efficiency of the emergency core cooling system the guidelines include the requirements (section 22.1.1) on the maximum fuel rod cladding temperature (1200 °C), local oxidation depth as a fraction of actual cladding tube wall thickness (17 %) and the maximum fraction of zirconium of all cladding tubes reacting with water (1 %), which are identical to the corresponding requirements laid down in the United States Code of Federal Regulations, Title 10, Section 50 (10 CFR 50) appendix K "ECC evaluation models". Beyond these criteria the guidelines request that specified fission product releases will not be exceeded as a result of cladding tube defects. An additional decoupling criterion for core damage extent analysis has been established: the fuel rod cladding failure shall not exceed 10 % of II rods. Further requirements are that the geometry of the reactor core shall not be subjected to changes which prevent its sufficient cooling, that the subcriticality shall also be assured in the long term range without considering the effectiveness of the scram system in the long term reactivity balance. In the analysis of operational transients it has to be demonstrated according to section 3.1.3 of the RSK Guidelines that the heat flux is sufficiently far from the critical heat flux, that the pressure within the pressure retaining boundary remains below the response pressure of the pressuriser safety valves, and that the energy release inside the fuel rods is so low that no melting occurs. No opening of a pressuriser relief valve is permitted for frequent transients.
The RSK guidelines require in section 22.1.3 that the analyses must be experimentally confirmed and request assumptions for the thermal hydraulic calculations. The licensing authorities have therefore requested that the code users demonstrate a sufficient degree of validation of the applied codes. Latest experimental results are requested as the basis for models in the codes. For the case that the experimental evidence is not available or cannot be submitted in the licensing procedure the use of such correlations is requested, which are considered as conservative by the commission. The assumptions for the code in section 22.1.3 (also in 22.1.2 and 21.1) comprise:

1. **Blowdown Rate**: experimentally confirmed relation or suggested relations (Bernoulli/Moody), but conservatism to be justified with view to reactor core flow rate and most unfavourable effects upon fuel rod behaviour
2. **Burnout Delay**: equations for film boiling to be used after exceeding critical flux or correlations based on relevant experiment, conservative estimates for very large leaks
3. **Heat Transfer during blowdown and before dry-out of reactor core**: after exceeding the critical heat flux experimentally confirmed relations or modified Dougall-Rohsenow, it is to be checked whether other relations may lead to higher fuel rod temperatures in the hot channel
4. **Heat Transfer prior to flooding**: from end of blowdown (EOB) to bottom of core recovery (BOCREC) the heat transfer coefficients shall be based on experimental results, otherwise adiabatic core heat-up
5. **Leakage from the break**: water supplied to the point of break shall not be considered for core cooling
6. **Heat Transfer during quenching**: experimentally confirmed values, otherwise modified Dougall-Rohsenow equation
7. **Steam binding**: should be prevented, consequences for the practice (not written in the guide line): condensation effects of hot leg injected water challenges the code calculation, following the spirit of the guide lines: experimentally confirmed correlations or conservative assumptions.
8. **MCP behaviour**: experimentally determined variables or conservative assumptions
9. **Residual water content**: no residual water at EOB, other assumptions have to justified and confirmed by conservative calculations
10. **Flow rate reduction**: for hot channel calculation 20% reduction of mass flow rate obtained from the one-dimensional calculation due to flow distribution caused by thermal hydraulic influences or experimentally confirmed values
11. **Back pressure in containment**: reduction of 20% of analysed containment pressure
12. **Core power distribution**: most unfavourable values which may occur during specified normal operation under consideration of the limiting features (e.g. peak value of reactor limitation system RELEB)
13. **Pressure differences in RPV**: adequate subdivision into zones for its determination
14. **Decay heat power of the core**: ANS Standard + 20% (DIN 25463 + 2s was accepted in the licensing of KONVOI plants)
15. **Long term steam release**: on a long term basis following the end of the quench-
Long term NPSH of LPIS pumps: after change over to sump operation atmospheric pressure in the containment shall be assumed as far as NPSH of the LP injection pumps of the ECCS is concerned.

In section 22.1.2 the relevant assumptions are:

17 Failures in other safety systems than ECCS: components and systems e.g. emergency feedwater pumps, secondary cooldown valves including their activation, required in addition to the ECCS or instead of ECCS to cope with accidents (e.g. transients, SBLOCA) shall be considered like ECCS with regard to the requirements, e.g. failures due to single failure and maintenance.

Another important example is the assumption of the most effective control assembly withdrawn (stuck rod). The stuck rod can be assumed as the single failure, in several analysis the stuck rod was also assumed as failure independent of the single failure.

18 Loss of Power: shall be postulated for the DBA

19 Secondary Depressurisation: may only be considered in accident analyses if automatically initiated and performed, otherwise like all manually initiated systems after a grace time of 30 min.

20 Flooding tanks: available water supply shall be sufficiently conservative (relevant if there is no sump operation of HPIS pumps)

and in sections 21.1 and 21.2:

21 Postulated leak cross sections in main coolant pipes (MCP):
- reaction and jet forces on pipes, components and its internals, on buildings:
  0.1 A (A = open cross section) with linear opening time of 15 ms
- efficiency of ECCS: up to double-ended guillotine break (DEGB) of the main coolant pipe with linear opening time of 15 ms
- containment vessel design and incident resistant electrical equipment:
  up to DEGB of MCP
- stability of large components (RPV, SG, MCP, PRZ): \( p \cdot A \cdot S \) with \( p = \) operating pressure, \( A = \) open cross section, \( S = 2 \) (safety margin)

22 Postulated leak cross section in reactor pressure vessel:
- 20 cm² below core upper edge (larger leaks shall be detectable by suitable monitoring measures)
- break of a control assembly nozzle

23 Postulated leaks and breaks in the main steam and feedwater pipe:
- with regard to the stresses acting on SG tubes: whole spectrum of leaks and breaks including the inadvertent opening of valves, however in this incident analysis the failure of a few tubes shall be postulated as single failure by assuming a DEGB of one tube in the concerned SG
- with regard of the effects of overcooling transients on the reactivity behaviour and on the changes in pressure and temperature in the reactor as well as the resulting stresses on the RPV and its internals

and in section 3.1.2/3.1.3:

24 Scram activation by different process variables:
As a matter of principle the scram release is to be activated by different process variables which are redundant among themselves (section 3.1.2). As a consequence the first scram activation signal in the accident analysis is ignored. Ex-
ceptions are possible for certain incidents when activation can only originate from one process variable. In these cases a higher safety level in the design of the detection and transmission of the corresponding signal must be assured.
In section 3.1.3 (2) the failure of the first scram activation is explicitly requested for operational transients requiring reactor scram.

It is obvious from the decoupling criteria listed above together with the corresponding explanations and comments that the deterministic thermal-hydraulic analyses were performed under conservative and best estimate conditions in all licensing procedures in Germany. It was never a request in the rules and guidelines to apply a conservative evaluation code with licensed assumptions in the licensing procedure. This kind of procedure allows to flexibly follow the advances in safety technology and to transfer reliable results of research and development into code models and assumptions. A sufficient part of conservatism is always ensured by means of deterministic postulates. As long as realistic models were not available conservative assumptions had to be applied. Within this flexible kind of procedure unnecessary conservatism can be abolished if no longer needed after incomplete knowledge as the reason for introducing the conservative postulates is replaced by sound knowledge. Some examples shall illustrate these changes.

1.3 Examples for the Development Towards Best Estimate Approach

After numerous system experiments (e.g. in LOFT, PKL) of different scaling a database was created for code validation. One of the important findings was the existence of residual water after the depressurisation phase (at EOB). Consequently the assumption of no residual water (number 9 of the above list) was no longer justified. After validation of the involved code models, in particular the models for interfacial friction (e.g. drift flux models) and for counter current flow limitation in the downcomer geometry, the codes can predict a best estimate value of the residual water. The experiments came too late to influence the design of the NPPs. Otherwise the initial pressure of the accumulators could have been raised up. The selection of the relative low pressure of the accumulators was connected to the intention to still have the majority of the initially stored water available for the injection after the EOB. A higher pressure would have had several advantages which could not be used due to the requirement at the time of the design and fabrication.
A second example for changing an analysis assumption suggested in the RSK guidelines concerns the heat transfer correlation after exceeding the critical heat flux. Code validation analyses of several separate effects tests did not confirm the conservatism of the modified Dougal-Rohsenow correlation suggested as a conservative bound in the guidelines (numbers 3 and 4 of the above list). Other correlations were found to be conservative, e.g. the Groeneveld 5.9 correlation was following the data trend better still being on the conservative side in all ranges of application. In the licensing analysis therefore this correlation was applied.

The third example is not related to one of the assumptions requested by the RSK guidelines but it concerns the general development of the thermal hydraulic system codes. Until 1984, a conventional drift-flux model was used in the codes for all essential flow regimes in vertical geometry, supplemented by correlations for the counter-current flow limitation. For horizontal flow channels, only simplified void-correlation were available. Validation was performed by means of available experiments from the test facilities LOBI, BATTELLE and LOFT. In order to compensate the model uncertainties discovered during this validation, conservative model approaches with regard to the counter-current flow limitation were made for the reactor calculations for the KONVOI plants. Conservatism in this case means that water flow down from the steam generators to the upper plenum was hindered in the horizontal hot leg by the steam flow in the opposite direction. The results obtained with such models complied with the state of the art at that time, as comparisons of results from different codes (RELAP and DRUFAN) showed. After 1984 experimental results from the UPTF test series had been submitted, a new correlations related to the geometry for the horizontal tube which also includes the inclined tube were developed using limit values for the counter-current flow limitation established by Bankoff and Lee for the horizontal tube. This correlation was verified by post-test calculations of the experiments performed by Richter and Ohnuki. The experiment UPTF 11 allowed for the verification of the correlation related to the counter-current flow in the horizontal tube. The geometric configuration of the test arrangement represented the hot train between reactor pressure vessel and steam generator in a KONVOI-type pressurised water reactor (Fig. 4). Saturated water was added from the steam-generator side and, simultaneously, saturated steam from the vessel side, i.e. from the upper plenum. The steam mass flow rate was increased gradually until occurrence of the counter-current flow limitation. The specific volumetric flow rates in terms of Wallis parameters for steam and water for the counter-current
flow limitation condition are presented in Fig. 5. For reasons of comparison, the analytical results achieved with the newly developed correlation by GRS are included in the figure. The good correspondence for the horizontal tube with original diameter demonstrates that the velocities of the two phases including the counter-current flow limitation are determined realistically by the correlation. With the new correlation, further experiments were analysed successfully, e. g. the depletion of the steam generator U-pipes during a small-leak experiment in the LOBI test facility. This experiment represented the international standard problem OECD-CSNI-ISP-18 (Fig. 6). After completion of the model validation calculations, the new correlation was incorporated in the ATHLET code.

In the same time period the designer had also considered the extended experimental data base and had improved - among others - the models for the two-phase counter-current flow in horizontal pipes. He also used the experimental results of UPTF 11 as the main contribution for the improvement of the corresponding models. He also arrived at good agreement and he implemented the model in the RELAP code.

For the analyses of small leaks in the frame of the licensing procedure these model improvements in the codes arrived too late for application. Hence the results remained conservative. In order to find out the degree of conservatism some of the licensing analyses were repeated with the improved code versions. A medium leak size of 160 cm² in the cold leg of the KONVOI plants was selected by the assessor, because this leak had lead to the highest cladding temperatures in the analyses performed in the frame of the licensing of the KONVOI plants. The applicant also repeated a similar leak size, a medium leak in the cold leg of 100 cm².

The results for the medium leak of 160 cm² in the cold leg of the main coolant line may serve as an example for it. In Fig. 7 and 8 comparison is made between the recent and former results. Fig. 7 shows the collapsed water level in the core. Fig. 8 illustrates the influences on the maximum cladding temperatures. The time sequence shown comprises the phase between incident occurrence and the beginning of the emergency cooling injection by the pressure accumulators. The high injection rate from the pressure accumulators terminated the uncovering of the core, which occurred in the former calculations. In the recent calculation, however, the core remains continuously covered with a two-phase mixture even during the high-pressure injection phase. As a conse-
quence, the cladding temperatures do not rise significantly above the saturation temperature of the coolant.

In the repeated analysis of the small leak with 160 cm², the time period during coolant is retained in the steam generators was reduced compared to the former calculation. With realistic simulation of the counter-current flow limitation the coolant entered the reactor pressure vessel earlier and caused an increase of the core water level. An uncovering of the core did not occur in the recent analysis. The same result was also obtained in the repeated RELAP analyses.

With the new correlation in the ATHLET code further analyses were performed by the technical inspection agency TÜV Hannover/Sachsen-Anhalt in 1992 or the leak spectrum of 50 cm² to 250 cm² during which core uncovering did not occur either. The KWU Analyses by the applicant (Siemens KWU), also performed in the years 1991/92 for the same leak spectrum, showed also no more core uncovering in contrast to the analyses performed in 1983.

In summary, it was proven that in the analyses on the small leaks, performed independently by the applicant and by the assessors in the frame of the licensing procedure for the KONVOI plants, the core heat-up had been overestimated due to a too strong coupling of the phases in the horizontal tube.

2 Current Approach

The thermal-hydraulic system codes presently available in Germany for licensing and related purposes are more realistic than at the time of licensing the KONVOI plants. Existing models were essentially improved, new models were developed, e.g. six equation two-fluid models, direct condensation models, models for non-condensable gases, boron tracking models, extended two-phase flow models for special components, e.g. T-junctions, complex balance-of-plant models. The robustness of the codes was essentially improved. System codes describing the thermal hydraulics have been successfully coupled with containment codes and with multidimensional codes describing the reactor physics. Based on a comprehensive validation including experimental results obtained in differently scaled test facilities up to full size facilities (e.g. UPTF) the models reflect the best available knowledge. Hence the first prerequisite
for the performance of best estimate analysis for nuclear reactors of present design is fulfilled. The application of these codes in the design and for the licensing of innovative reactors constitutes an issue of current interest.

In general, best estimate analyses is not identical with a realistic description of nature without any reservation. Some of the very complex phenomena can still not be described in a completely realistic manner mainly due to their multidimensional and micro-scale nature. The wide range of different condensation phenomena is an example for this. As long as reliable two-phase CFD codes with turbulent diffusion models are not available for a more realistic description of such phenomena and as long as the results of a comprehensive uncertainty evaluation for the present codes are not available, the parameters of specific models will be set to conservative values.

High qualification of code users is the second prerequisite for best estimate analysis. Training and practical experience preferably in code validation activities is of high importance. User manuals are available for all major codes which assist the users in the effective application of the codes. Practical instructions and guidelines are given in the manuals that enable the user to compile input data, to perform calculations and to evaluate the results. Recommendations for an appropriate nodalisation are given. Nevertheless the influence of the user on the result of code calculations can still be large. As numerous national and international standard problems have proven, it cannot always be achieved that two or more users having available the same code and the same information for developing a nodalisation and running the calculation do not arrive always at the same results. The best and very similar results were obtained from those users of different codes which had the best experience due to their continuous involvement in code validation. It is presently one of the major efforts to reduce the user effects. One possibility is the reduction of model options in the codes. This possibility is limited by the required flexibility of code application for different reactor systems. The German practice that applicant and assessor apply in general different codes for the same case is very effective in identifying user mistakes as well as code errors provided a technical discussion between the code users is carried out. A detailed discussion is very helpful to discover even compensating effects.

The third prerequisite for best-estimate analysis is an evaluation of uncertainties. In parallel to the trend towards more realistic best estimate calculations, methods for the quantification of uncertainties of the calculated results were developed, tested and
partly applied. Binding regulations for the uncertainty quantification do not exist yet in Germany. The discussion in the French and German advisory committee is in full activity. At present RSK together with GPR, the French advisory committee, prefer the use of realistic assumptions and models for the safety demonstration under the condition that the compliance of the results with the acceptance criteria is proven at a high confidence level which means also the explicit evaluation of the associated uncertainties. However, the committees do not recommend it as the only alternative. They would also tolerate the use of models and criteria in the conservative approach as applied in the past. This alternative, however, is not preferable because the high level of code validation could not be utilised in this case. Also conservative hypotheses may not always lead to really conservative results. An example is the conservative increase of the reactor power which may lead to an overprediction of the core swell level. The resulting overprediction of core cooling would be opposite to the intended conservative results.

Two methods for the quantification of the uncertainties are available at present in Germany. The designer's method follows essentially the CSAU approach, but differs in the application of some steps. It is summarised below. More insights and details about this methodology are presented by F. Depisch from Siemens in Session 3 of this seminar. The GRS method will be applied for the confirmatory analyses conducted as part of the licensing process by the expert organisations. This statistical method is also summarised below. More details are presented in Session 3 of this seminar by H. Glaeser.

The present approach of the designer follows the CSAU approach but differs somewhat in the application. The methodology had been developed by U.S. Siemens Power Corporation (SPC) in 1993 and has been submitted to the US NRC for approval. It was recently (1996/97) adapted by Siemens for a 4 loop PWR 1300 of KWU type. The approach consists of 3 parts. In the first part the Phenomena Identification and Ranking Table (PIRT) is established for each selected event sequence, e.g. for LBLOCA of a NPP. Within the second part the specific assessment of the applied code related to the selected event sequence is demonstrated. A cross reference matrix contains the experiments (separate effects tests and integral tests) used for the assessment of the phenomena of the PIRT. The validation results are documented also as part of the second part. The plant model (nodalisation) for the analyses of the selected event sequence is established in consistence with the analyses of the experiments. A sensitivity
analysis is performed within the third part. Based on these results a response surface is established. Separate uncertainties are grouped into three basic categories resulting from code parameters, plant parameters, e.g. the initial conditions, and fuel parameters, e.g. power, total peaking factor, gap conductance, and decay heat. Using the response surface the separate uncertainties of selected criteria, e.g. the peak cladding temperature (PCT) for LBLOCA, are combined via the Monte Carlo Method by means of performing a large number of calculations. Finally the results of the statistical analysis is presented for the selected parameters and for a selected confidence level, e.g. an upper limit of the PCT for 95 % confidence level and with a probability of 95 %.

GRS completed the development of the GRS method, applied it on experiments and performed a first reactor analysis. Work is going on to improve the knowledge on highly sensitive input parameters. At present GRS is prepared to apply their method using simplified conservative uncertain input parameter distributions for those variables for which the data base is not yet validated sufficiently. Like for the designers approach, the validation experience is also for the GRS method the basis for the quantification of parameter uncertainties. An important experience is that the agreement of calculated results with experimental values was often obtained by changing parameter values for different experiments in the same facility and for similar experiments in different test facilities. The uncertainty of the calculated results is influenced by uncertainties of input data (e.g. models and model parameters, material properties, nodalisation, numerical parameters, e.g. convergence criteria, scaling effects and others). Also the measurement errors of experimental data influence the uncertainty. The aim of the uncertainty analysis is at first to identify and quantify all potentially important uncertain parameters. The state of the knowledge about uncertain parameters is described by subjective probability distribution or by their probability density function. The combination of the uncertainties is achieved by a simultaneous variation of a random sample of sets of uncertain parameter values. This is different from the Siemens approach. The required number of thermal hydraulic computer code calculations for the statistical evaluation by means of the code SUSA (Software System for Uncertainty and Sensitivity Analysis) only depends on the desired statistical tolerance limits and not on the number of input and output parameters (PCT, water inventory, pressure, core pressure differences etc.) of the uncertainty analysis. As an example the minimum number of code runs for limits not to be exceeded (one sided tolerance limit), e.g. PCT, with a probability of 95 % depends on the required confidence level. For 95 %
confidence level 59 code runs are the minimum, for smaller confidence level a smaller number of code runs, e.g. 45 runs for 90 %. The tool SUSA allows also to avoid unphysical combinations of input parameters within the specified uncertainty of input parameters. The results of the thermal hydraulic calculations are directly used without response surface or other approximations. The GRS method was at first applied for a separate effects test (OMEGA) and an integral test (LSTF) and thereafter for a SBLOCA in the PWR-1300.

The final result is the quantification of the uncertainty limits of the calculated results representing the combining influence of all parameter uncertainties. The method also serves for the determination of the major sources of uncertainty of calculated results (ranking of uncertain parameters), enables to determine the margins to safety limits, guides further code development and identifies priorities for further code development.

3 Future Approach

The future approach is characterised by the further development of the present approach. Conservative assumptions for compensation of incomplete knowledge about model uncertainties will disappear. Each analytical result will be accompanied by a statement of its uncertainty. A detailed uncertainty analysis may not always be necessary for results which are very far from the limits. The presently by French and German advisory committees developed guidelines for future reactors will consider the evaluation of uncertainty as a very important issue. Combined uncertainty evaluation will be performed for thermal hydraulics and fuel rod behaviour. The initial and boundary conditions will be included in the uncertainty evaluation.
Fig. 1: Nuclear Licensing Procedure in Germany

Fig. 2: Legal basis of the peaceful utilisation of nuclear energy in Germany
- Date of Application
- Date of Commercial Operation

**Fig. 3:** Operating nuclear power plants in Germany
Fig. 4: Schematic representation of the hot leg of UPTF

Fig. 5: Comparison of experimental results (UPTF 11) and analytical results with drift flux model of GRS, in terms of square root of Wallis parameters for both phases
Fig. 6: Δp over U-tubes of steam generator (proportional to water level) of OECD-CSNI ISP-18 versus GRS analysis result (after implementing the advanced drift-flux model)

Fig. 7: Application of the advanced drift-flux model: SBLOCA in cold leg of PWR-1300 on collapsed level in core
(previous analysis: conventional model, recent analysis: advanced model)
**Fig. 8:** Application of the advanced drift-flux model: SBLOCA in cold leg of PWR-1300 on hot rod cladding temperature.

(previous analysis: conventional model, recent analysis: advanced model)
APPLICANTS ACCIDENT ANALYSES IN SECOND PART LICENSE
FOR KONVOI-PLANTS

Safety Analysis Report (S.A.R.) with 3 Handbooks:
- ECC Handbook (LOCA)
- Plant Dynamics Handbook (Transients incl. ATWS)
- Core Design Handbook

Content of ECC-Handbook:

0 Purpose, Scope, Handling, Abbreviations

Living handbook, Basis for design, catalogue of transients, specifications and licensing.
Handbook contains LOCA in primary system, it contains also core damage analysis,
description of codes, description of essential plant data and code input data.

1 Initial and boundary conditions

1.1 Description of Essential Systems

Emergency core cooling and decay heat removal system, systems for secondary heat
removal.

1.2 Reactor Protection System

Initiation criteria for scram, MCP shut-off, MFW shut-off, closure of MSIVs, for ECCS,
emergency power diesel load sequence, accumulator shut-off, automatic secondary
cooldown, secondary side safety valve isolation valve shut-off.

1.3 Reactor Core
Initial and boundary core conditions, kinetics data

1.4 Code-specific Boundary Conditions

Codes for LBLOCA (2A-0.25A): LECK-4/MOD2 (ANLASS = stationary conditions, BENN = leak mass flow rate, LECK = system code for instationary blowdown phases), HYDRANS = injection mass flow rate, REWAS = residual water in lower plenum, WAK-3 = refill and reflood, BETHY-AZ = maximum cladding temperatures, COCO = containment back pressure, CARO-D = initial conditions of fuel rods, Code for SBLOCA: RELAP5/MOD1

1.4.1 LECK-4/MOD2

boundary conditions, nodalisation

1.4.2 HYDRANS

boundary conditions, characteristics, nodalisation

1.4.3 WAK-3

boundary conditions

1.4.4 BETHY

boundary conditions

1.4.5 RELAP5/MOD1

initial and boundary conditions, nodalisation

1.5 Relevant Plant Differences between Pre-KONVOI (Generation 3: KBR, KWG, KKP 2) and PWR 1300 (Konvoi)

main differences: fuel elements, MCP, lower plenum internals
2 Summary of Results of LBLOCA ECC Calculations

7 LBLOCA calculations: 2A in CL, PCL, HL, 1A in CL, 0.5 A in CL, 0.25 A in CL, 2A in CL (best estimate)

3 Results of LBLOCA ECC Calculations

General description of LBLOCA sequence

3.1 Double-Ended Guillotine Breaks (2A)

3.1.1 2A in Cold Leg

Conservative initial and boundary conditions (e.g. SF and repair)

3.1.2 2A in Crossover Leg

3.1.3 2A in Hot Leg

3.2 Leak Size Less than 2A

3.2.1 1A in Cold Leg

3.2.2 0.5A in Cold Leg

3.2.3 0.25A in Cold Leg

3.3 Special Investigations

3.3.1 2A in Cold Leg with best estimate conditions

Purpose: Identification of safety margins, all systems available, no emergency power conditions,...

4 Results of ECC Calculations for SBLOCA and MBLOCA

General description of LBLOCA sequence, definition of SBLOCA, MBLOCA

4.1 Leaks of Main Coolant Lines
All individual reports contain description of events, initial and boundary conditions, assumptions on availability of ECCS, calculated results, nodalisation scheme, about 40 plots

4.1.1 1105 cm$^2$-leak (0.25A) in cold leg
4.1.2 380 cm$^2$-leak in cold leg (injection line)
4.1.3 380 cm$^2$-leak in hot leg (injection line)
4.1.4 437 cm$^2$-leak in hot leg (surge line)
4.1.5 250 cm$^2$-leak in cold leg
4.1.6 160 cm$^2$-leak in cold leg
4.1.7 100 cm$^2$-leak in cold leg
4.1.8 70 cm$^2$-leak in cold leg
4.1.9 50 cm$^2$-leak in cold leg

4.2 Leaks at Reactor Pressure Vessel

Fulfilment of RSK guide line

4.2.1 20 cm$^2$-leak at bottom

4.3 Leaks at pressuriser

4.3.1 Inadvertent opening and stuck-open of relief valve (20.8 cm$^2$-leak)

5 Results of core damage analysis

5.1 Core damage analysis

5.1.1 Realistic and conservative core damage

5.1.2 Influence of break location and size on core damage
6 Description of codes

6.1 LBLOCA

6.1.1 Blowdown phase (LECK-4/MOD2 code system)

Users manual, input data, output data.

6.1.2 Refill and reflood phase (HYDRANS, WAK-3, REWAS)

Users manual, input data, output data, verification based on PKL test K9

6.1.3 Core heat-up (BETHY-AZ)

Model description, input deck

6.2 Small and medium leaks

6.2.1 RELAP-5/MOD1

Code manual (system models, numerical methods, user guide, input requirements)

6.2.2 BETHY

see 6.1.3

6.3 Core Damage Analysis

6.3.1 BETHY-AZ

7 INPUT DATA

Complete input decks for each case

7.1 LBLOCA

7.2 SBLOCA and MBLOCA

7.3 Core Damage Analysis
Content of Plant Dynamics Handbook:

0 Purpose, Scope, Handling

Basis for design, commissioning, operation, catalogue of transients, specifications and licensing
Specified operation, Disturbed operation, incidents, not: LOCA, SS-procedures
Code description

1 Analyses

1.1 Specified Operation

1.1.1 Normal Operation

(analysed for KKP-2, because of same controllers for coolant pressure and temperature, Code LOOP-7)

1.1.1.1 Slow load changes (+- 10%/min)
1.1.1.2 Fast load changes (+- 10%)

1.1.2 Abnormal Operation

1.1.2.1 Turbine shutdown

(Code: NLOOP, Calculation for BOC and EOC)

1.1.2.2 Load Rejection (covered by 1.1.2.1)

1.1.2.3 Fast reactor shut down (scram)

(Code: NLOOP, Calculation for EOC, FP, Scram initiates turbine shutdown, i.e. PCS coolant temperature falls only to secondary saturation temperature)

1.1.2.4 Failure of one MCP

(Code: NLOOP, Calculation for FP at BOC and EOC, no scram, new power level about 40%)
1.1.2.5 Failed function of auxiliary systems in PCS

(no code application, failed function of CVCS: coolant mass, coolant temperature, boron concentration)

1.2 Disturbed Operation and Incidents

1.2.1 Reactivity and Power Distribution Anomalies

(analysed for KKP-2 except for 1.2.1.4.2 because of same effectivity of control elements and because of conservatism of the analyses, but KKP-2 = 16x16, Convoy = 18x18)

(in common: Code LOOP-7 with the exception of 1.2.1.4.2, ignorance of first scram signal and of all controllers for control rod movement)

1.2.1.1 Incidents during plant start-up

(ZL, 225°C, disturbance 60 pcm/s, 20 pcm/s due to erroneous withdrawal of control elements with 1.25 cm/s)

1.2.1.2 Insertion of control elements

(covered by 1.2.1.3)

1.2.1.3 Drop of one control elements

(Δρ=-0.2%)  

1.2.1.4 Withdrawal of control elements

1. \( \frac{dp}{dt}=2.875 \text{pcm/s}, \) ignorance of PBCORR as first scram signal, second: PK

2 Withdrawal of D-Bank with 15 pcm/s: 4 calculations with code NLOOP (BOC and EOC from 50% initial power, BOC and EOC from FP)

1.2.1.5 Ejection of one control element

(Δρ=+0.2%, ignorance of PBCORR as first scram signal, second: PK)

1.2.1.6 Change of boron concentration due to failed function of CVCS
(dp/dt=0.7pcm/s, ignorance of PBCORR as first scram signal)

1.2.1.7 Detaching of boron containing sediments in core

(Δp=+0.25%, ignorance of PBCORR as first scram signal)

1.2.2 Disturbed Heat Removal without Loss of Primary Coolant

1.2.2.1 Malfunction of turbine bypass valves

(Code: NLOOP, 3 cases: Opening of one bypass valve, malfunction of all bypass valves with and without RELEB)

1.2.2.2 Fast turbine shutdown with failures of various systems

(Code: NLOOP, several cases at FP: failure of condenser at BOC and EOC, same but ignorance of first scram signal at BOC, turbine shutdown without turbine bypass at BOC and EOC, same but ignorance of first scram signal at BOC)

1.2.2.3 LONOP

(Code NLOOP; FP, consideration SS pumps, of auxiliary spray from CVCS and of automatic partial cooldown if p>86 bar, scram due to n_{\text{MCi}}<94% and PBCORR at same time)

1.2.2.4 Erroneous closure of MSIVs

(Code: NLOOP; FP, SF: MS-SV, LONOP at time of scram; 5 cases: closure of one MSIV with first scram signal (p_{\text{occ}} > 86 bar) with BOC and EOC, same two cases with second scram signal (SG level < min 1) but partial cooldown to 75 bar considered as in the cases 1 and 2, closure of all MSIVs with simultaneous scram)

1.2.2.5 Malfunction of FW supply

(Code: NLOOP; FP, 4 cases: failure of one MFW pump at BOC and EOC, failure of both MFW pumps at BOC and EOC.)

1.2.2.6 Blockage of one MCP

(Code: NLOOP, several cases with two different time constants: at BOC with and without consideration of RELEB and at EOC)
1.2.3 Incidents with Loss of Main Steam

1.2.3.1 Inadvertent opening of one safety valve

(Code: NLOOP, several cases at ZL (no decay heat) and at FP (FP: 900 ppm, EOC: 0 ppm)), at p < 60 bar safety valve isolation valve closes)

1.2.3.2 Break of one main steam line (MSL)

see 1.2.3.4

1.2.3.3 Break of one main feed water line (MFWL)

(no analysis, but argumentation: break upstream of one-way valve is covered by total failure of main feed water, break between one-way valve and steam generator is covered by leak in MSL)

1.2.3.4 Leaks in MSL or MFWL

(Code: NLOOP, calculations for FP, ZL (no decay heat), EOC without boron, analyses of following cases: DEGB of MSL outside containment, 0.1A leak in MSL inside containment, to increase the recriticality at leak inside containment for ZL, the additional failure of safety injection into defect loop was assumed as a further case)

1.2.4 Incidents with Loss of Primary Coolant

1.2.4.1 Steam generator tube rupture

(Code: NLOOP; FP, 2A SGTR, EOC, consideration of all automatic actions derived from N16-signal, no operator actions before 30 min.; 2 cases: with and without loss of power at the turbine shutdown after scram due to PCS pressure)

1.2.4.2 Inadvertent opening of pressuriser valves

(Code: NLOOP, two cases at FP, BOL: without and with ignorance of 1st scram signal (p<132 bar), second signal: level in pressuriser > 9.5m)

1.2.5 Incidents with Loss of Main Steam and SGTR

1.2.5.1 Break of MSL and SGTR
(Code: NLOOP; FP, EOC without boron, SGTR assumed as an additional, independent, random failure, considered as SF, no repair case because of small probability event, no operator actions before 30 min.)

1.2.5.2 Stuck open of MS Safety Valve and SGTR

(Code: NLOOP; FP, SGTR assumed as SF, no operator actions before 30 min.)

1.3 Anticipated Transient without Scram (ATWS)

Analysis of 8 ATWS cases, i.e. failure of the scram system during operational transients, according to RSK guideline 20. For each case two different calculation were performed: failure of the signalization for reactor scram, mechanical blockage of all control elements. Generally BOC (960 ppm), for subcooling transients also EOC assumptions.

All analyses were performed originally for KKP-2. (Differences small, comparatative calculations were performed.

1.3.1 Failure of Main Heat Sink

Possible reasons: loss of condenser vacuum, closing of MSIV

1.3.2 Failure of Main Heat Sink + LONOP (station service power supply)

1.3.3 Maximum Increase in Steam Extraction

Possible reasons: opening the turbine bypass station, opening of main steam safety valves. Calculations for BOC and EOC.

1.3.4 Maximum Reduction of Feedwater Supply

Possible reasons: Failure of one MFW pump (reserve pump not available), Failure of both MFW pumps (also reserve pump not available).

1.3.5 Maximum Reduction of Coolant Flow Rate

Failure of one MCP
1.3.6 Maximum Reactivity Gain

Possible reasons: Withdrawal of control assemblies or control assembly banks

1.3.7 Primary System Depressurisation

Possible reason: Unintentional opening of pressuriser safety valve

1.3.8 Maximum reduction of reactor inlet temperature

Possible reason: Malfunction of an active component of feedwater supply system, e.g. HP-preheater

2 Description of Computer Codes

NLOOP combines thermal hydraulic and I&C simulation. Fixed zones in PCS, energy and mass exchange with other components (pressuriser, SG, RPV-upper head,...), integrated momentum balance, 1 - 4 loops, one-dimensional in all components, point kinetics, homogeneous coolant, in general thermal equilibrium (exceptions are in pressuriser, SG,...), mixing in lower plenum, change of flow direction possible, average fuel rod per loop, radial heat conduction, decay heat (DIN), reactivity due to void, boron, spectral, Doppler, control rod, pressuriser: heater, spray, relief, relief tank, SG: level, average tube, RPV upper head steam bubble homogeneous, MS-system: quasi-stationary compressible model, SGTR simulation, MSL leak, all limitation systems, control systems, reactor protection system.

Verification by means KKP-2 commissioning test, e.g. shut-off of one MFW-pump and switch on of one AFW-pump.

NLOOP is an extended version of LOOP-7. Main difference: extended simulation capability of NLOOP (1-4 loops) compared with LOOP-7 (only one loop).

3 Reference Material and Data
Content of core design handbook:

0 Introduction

1 General fuel assembly and core data
Description of reactor core geometry, the reactor core loading map and the Fuel assembly design

2 Nuclear Design
Monitoring of power density and power density distribution.
Reactivity balance and reactivity coefficients, efficiency of shutdown systems.
Calculation of burn up cycle, power density distribution, critical boron concentration.
Codes used: SAV79A standard analysis methodology including FASER for nuclear data generation,
MEDIUM and PANBOX for static and transient core calculations.

3 Thermal-hydraulic Design
Collection of relevant design data.

4 Description of first Core Loading

5 Accident analysis
- Startup accident
  Maximum reactivity insertion by withdrawal of control rods
- Withdrawal of a single control rod or a group of control rods
  Core conditions: BOC, EOC, hot zero power, full power
- Rod ejection accident
  Core conditions: BOC, EOC, hot zero power, full power
Code for core transients: PANBOX, 3D coarse mesh solution of neutron diffusion equations by nodal expansion method.
ASSESSOR ACCIDENT ANALYSES IN SECOND PART LICENSE FOR KONVOI-PLANTS

The three TÜV (Technical Inspection Agencies) responsible for the three individual plants of type KONVOI: TÜV Bayern for ISAR-2, TÜV-Hannover for KKE, TÜV-Stuttgart for GKN-2 and GRS performed the safety assessment. The work was distributed as follows:

- TÜV-Bayern for disturbance and failure of secondary heat sink without loss of coolant (failure of main heat sink, erroneous operation of valves in MS and in FW system, failure of MFW supply), long term LONOP, performance of selected SBLOCA analyses,

- TÜV Hannover for disturbances due to failure of MCPs, short term LONOP, damages of SG tubes incl. SGTR, performance of selected LOCA analyses (blowdown phase of LBLOCA),

- TÜV-Stuttgart for breaks and leaks in MS and FW system with and without leaks in SG tubes,

- GRS for ATWS, subcooling transients due to disturbances on secondary side, initial and boundary conditions for transients with opening of pressuriser valves with and without stuck-open, most of the LOCA analyses

---------------------------------------------

Analyses Performed in the Framework of Assessing the vendors Calculations in the ECC Handbook (LOCA)

1 LBLOCA

1.1 Analysis of 2A DEGB in Cold Leg with DRUFAN/FLUT

(analysed for KONVOI by TÜV-Hannover and by GRS)
ASSESSOR ACCIDENT ANALYSES IN SECOND PART LICENSE FOR KONVOI-PLANTS

The three TÜV (Technical Inspection Agencies) responsible for the three individual plants of type KONVOI: TÜV Bayern for ISAR-2, TÜV-Hannover for KKE, TÜV-Stuttgart for GKN-2 and GRS performed the safety assessment. The work was distributed as follows:

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Analyses Performed in the Framework of Assessing the vendors Calculations in the ECC Handbook (LOCA)

1 LBLOCA

1.1 Analysis of 2A DEGB in Cold Leg with DRUFAN/FLUT

(analysed for KONVOI by TÜV-Hannover and by GRS)
1.2 Analysis of 2A DEGB in Cold Leg with TRAC-PF1

(analysed for KBR by GRS)

1.3 Analysis of Hot Rod Analysis and Core Damage for 2A DEGB in Cold Leg

(analysed with TESPA by GRS, TESPA = probabilistic investigations considering distributions of decay heat, gap conductance, internal fuel rod pressure, criterion for fuel rod burst, thermal hydraulic boundary conditions)

2 SBLOCA AND MBLOCA

2.1 Analyses with DRUFAN

2.1.1 Analyses for KONVOI

2.1.1.1 800 cm² in cold leg

(Δ-Analysis Convoy/KKP-2)

2.1.1.2 160 cm² in cold leg

(largest cladding temperature in GRS analysis)

2.1.1.3 20 cm² at pressuriser

(entrainment in pressuriser)

2.1.2 Analyses for NPP KKP-2

2.1.2.1 380 cm² in cold leg

(rupture of largest connecting pipe at cold leg)

2.1.2.2 800 cm² in cold leg

(Δ-Analysis Convoy/KKP-2)
2.1.2.3 70 cm² in cold leg

(check of vendor's statement that no core uncovery takes place for leaks <100 cm²)

2.1.2.4 380 cm² in hot leg

(relevant case with regard to systems failure assumptions)

2.1.3 Analyses for NPP KWG

2.1.3.1 380 cm² in cold leg

(rupture of largest connecting pipe at cold leg, Δ-Analysis KONVOI/Grohnde)

2.1.3.2 70 cm² in cold leg

(check of vendor's statement that no core uncovery takes place for leaks <100 cm², Δ-
Analysis KONVOI/Grohnde)

2.2 Analyses with RELAP5/MOD1

2.2.1 Analyses for Convoy

2.2.1.1 380 cm² in cold leg

(rupture of largest connecting pipe at cold leg)

2.2.2 Analyses for KKP-2

2.2.2.1 70 cm² in cold leg
Analyses Performed in the Framework of Assessing the Vendors Calculations in the Plant Dynamics Handbook

1 Abnormal Operation

1.1 Failure of Main Heat Sink
(Code: RELAP4/2STD)

1.2 Failure and Restart of a MCP
(Code: RELAP4/2STD)

2 Disturbed Operation and Incidents

2.1 Ejection of one Control Element
(Code: RELAP4/2STD)

2.2 LONOP
(Code: RELAP4/2STD)

2.3 Closure of all MSIVs with Add. Failure of Depressurisation in one Loop
(Code: RELAP4/2STD)

2.4 Failure of MFW Supply
(Code: RELAP4/2STD)

2.5 Non-Isolable 2A MFW Line Break
(Code: RELAP4/2STD)
2.6 SGTR and erroneous opening of Relief Valve in Defect Loop

(Code: RELAP4/2STD)

3 ATWS

(Code: ALMOD 3.5, Selection of 3 worst cases from vendors calculations with the assumption of mechanical blockage of control elements, BOC, each 2 calculations: with KONVOI-specific reactivity feedback and with KKP-2-specific reactivity feedback)

3.1 Emergency Power Case

3.2 Failure of Main Feedwater Supply

(parametric study on the influence of steam generator heat transfer)

3.3 Failure of Main Heat Sink
Analyses Performed in the Framework of Assessing the Vendors Calculations in the Core Design Handbook

1 Nuclear Design

Calculation of fuel assembly characteristics like local pin power distribution and reactivity dependence of fuel assembly during burn-up cycle. Determination of homogenised two group cross-sections. Code: CASMO

Calculation of power density distribution in the reactor core and shutdown reactivity. Reactivity efficiency of boron and moderator temperature reactivity coefficient for a low-leakage core loading in GKN-2.

Code: QUABOX/CUBBOX-HYCA

2 Reactivity Initiated Transients

Check of initial conditions and assumptions on reactivity insertion in vendor calculations. Additionally, calculations by a model like ALMOD.

3 Steam-line break analysis

Calculation of reactivity balance and power density distribution during cooldown in shutdown condition including a stuck-rod by static calculations with QUABOX/CUBBOX-HYCA.
SESSION II:
OVERVIEW OF PAST AND CURRENT CSNI THERMAL-HYDRAULIC ACTIVITIES
ACTIVITIES OF PRINCIPAL WORKING GROUP 2 ON COOLANT SYSTEM BEHAVIOR

M.REOCREUX : Chairman

OECD/CSNI SEMINAR ON BEST ESTIMATE METHODS IN THERMALHYDRAULIC SAFETY ANALYSIS
Ankara, Turkey, 29 June-1 July, 1998-
GENERAL MANDATE OF PWG2

☐ SAFETY DOMAIN COVERED BY PWG2:

- Phenomena taking place within the reactor coolant system (in vessel)
- Whole spectrum of accidents from the most probable operational transients and accident to severe accident with extensive core damage

☐ ACCIDENT PHENOMENA COVERED BY PWG2

- In vessel Thermal-hydraulics during transients and accidents (LOCA, transients, EOPs)
- Core physics
- Fuel behavior
- Core degradation during severe accidents (thermomechanics, metallurgy, physics and chemistry, thermalhydraulics, fuel coolant interaction)

GENERAL MANDATE OF PWG2

☐ RELATION WITH OTHER CSNI WORKING GROUPS

◆ PWG4  Confinement of accidental radioactive releases
Counterpart of PWG2 in charge of ex vessel phenomena
Fission product, aerosols covered by PWG4 even for in vessel
FCI covered by PWG2 even for ex vessel

◆ PWG 1  Operating experience and human factors
possible interaction on simulators

◆ PWG3  Integrity of components and structures
interaction on the thermalhydraulics conditions which generate PTS

◆ PWG5  Risk assessment
interaction on accident evaluation, on accident management studies
ORGANISATION OF PWG2

□ PWG2  Leading group, one meeting per year, defines and co-ordinates the technical work performed by the task groups or task forces

Dr. Réocreux  Chairman
Dr. Aksan  Vice Chairman

□ TASK GROUPS  "Long term" technical groups

- Thermalhydraulic Applications TG-THA
- Degraded Core Cooling TG-DCC

Dr. Glaeser
Dr. Magallon

□ TASK FORCE  Technical group on specific problem for defined short or mid term action

- Task force on fuel safety criteria TFFSC

Dr. Van Doesburg

◆ WRITING GROUPS, AD HOC GROUPS, Sub groups of TG and TF for specific tasks

GENERAL TECHNICAL OBJECTIVES AND RELATED ACTIVITIES

◆ Forum for exchange of information between the most eminent experts in the fields

- Task group meetings
- Specialist meetings, workshops
- In general all the PWG2 activities

◆ Review of ongoing research, elaboration of synthesis

- State of the art reports, status reports
- Elaboration of validation matrices

◆ Detailed technical analysis

- International Standard Problems ISPs, Benchmarks

◆ investigation of new areas, elaboration of recommendations

- Status report
- Task group conclusions
INTERNATIONAL MEETINGS WITH VERY PARTICULAR SPECIFICITIES

- Restricted technical subject chosen because its interest is just in time
- Programme committee of highly involved technical people
  - Definition of a coherent programme (sessions, chairmen) for getting the wanted information
  - Large number of invited papers
  - Selected participation
- Elaboration of meeting summary and conclusions by the programme committee

MEETINGS OF VERY HIGH INTEREST

- PROCEEDINGS of high technical level
- DISCUSSIONS, PANELS the most interesting
- CONCLUSIONS valuable and largely based

STATE OF THE ART REPORTS, STATUS REPORT

- State of the art reports (SOARs) performed on selected subject where research results have reached some level of completion
- Status Reports (SRs), interim status on a selected subject of interest (safety issue) but where research is underway or should be started.
  - SOARs 2-3 years to be written, up to 300 pages with annexes
  - SRs less than 1 year to be written, some tenths of pages
- SOARs are unique reference documents which provide very valuable synthesis
- SRs are step reports which should help in the launching or in the orientation of researches
INTERNATIONAL STANDARD PROBLEMS (ISPs), BENCHMARKS

ISP's: Exercise commonly agreed consisting in

- calculating an experimental test offered by an host country.
- making the comparisons and the analysis of the different contributions
- drawing conclusions on the international capabilities of predicting the accident phenomena simulated in the chosen experiment

Well organised activity, Blind, semi Blind, Open exercise, format for experimental data,...

Certainly the most interesting activity organised by CSNI

- detailed technical exchange on the calculational tools
- common international technical experience and data base

Benchmarks: Similar to ISPs but without experimental reference

VALIDATION MATRICES

- Activity which was first started because of needs for code assessment (beginning of 80s)

- Selection of set of tests against which codes should be assessed in order to be applicable for plant accident calculations
  - Plant phenomena identification
  - Review of experiments and tests phenomena
  - selection of experiments and selection of tests

- Commonly agreed validation test matrix

- Data generally made available

- Central storage in a data bank.

- Activity started for Th codes. wide extension which has made from this activity a standard
  - extension to other codes ex: severe accident codes, containment codes,
  - extension to other types of reactors ex VVER
THERMALHYDRAULIC ACTIVITIES

The very first activity of CSNI in the 70s (ECCS working group)

SIGNIFICANT STEPS SINCE THE BEGINNING

☐ SEMISCALE TEST
  • LOCA DBA / code development and assessment, exp.pgm

☐ TMI2
  • Small break, transients, procedures / code devt and asst, exp pgm
  • Code validation / validation matrix
  • Best estimate codes application / Uncertainty methods

☐ "THERMALHYDRAULIC ISSUE IS CLOSED"
  • PWG2 continuing activities / val matrix, uncertainties, code devt and asst

THERMALHYDRAULIC ACTIVITIES

SIGNIFICANT STEPS SINCE THE BEGINNING (cont)

☐ SESAM GROUP
  • introduction of activities related to accident management / AM activities

☐ PRESENTLY 3 GROUPS OF ACTIVITIES
  • Th application / validation matrix, uncertainty methods, simulators
  • Th phenomenology / code devt and asst, exp pgm
  • Th accident management / reduced activity
THERMALHYDRAULIC ACTIVITIES RELATED TO PHENOMENOLOGY

☐ ISPs

The subjects of thermalhydraulic ISPs are following the safety issues concerns
- ISPs small break situations BETHSY, LOBI, LSTF, SPES
- ISPs for low power and shutdown conditions
- ISPs Boron dilution
- ISPs Advanced reactors
- PTS activity Thermal mixing
- BENCHMARK thermalhydraulics core physics (with NSC)
- REPORT lessons learned for ISPs on small break LOCA
- Revisit old ISPs

☐ WORKSHOPS, SPECIALIST MEETINGS

- Boron dilution Spec Mtg
- Annapolis Workshop
- Advanced instrumentation Spec Mtg

THERMALHYDRAULIC ACTIVITIES RELATED TO PHENOMENOLOGY

☐ STATE OF THE ART REPORTS AND STATUS REPORT

- SOAR BWR Stability
- SR thermalhydraulics aspects of advanced reactor designs
1. VALIDATION MATRIX

- 1 first matrix CSNI Code Validation Matrix (CCVM) independent assessment

- Now 2 matrices
  - Separate effect test matrix (SETM)
  - Integral test facilities matrix (ITFM)
    - SETM quite new (1995-96)
    - ITFM revision from CCVM issued 1996
    - Reference documents and reference approach
    - Data storage in NEA data bank

2. UNCERTAINTY EVALUATION

- Activity launched IN PWG2 since more than 10 years

- At the beginning considerable discussions on the basic principles of uncertainty evaluation methods (flowcharts,....)

- Comparisons of the different methods

- Discussion of the CSAU application on LB LOCA

- Some years without any significant progress (no significant research in the member countries

- Organisation of a Workshop

- Organisation of the Uncertainty Method Study (UMS)
  - 5 methods compared on a common calculation of LSTF SB CL 18 (ISP26)
  - Comparison report just published
ACTIVITIES ON THERMALHYDRAULIC APPLICATIONS

3. USER EFFECT STUDIES

- One of the most important difficulty in the uncertainty evaluation
- Very old story for PWG2 members. A product of ISPs.
- User effect with 1rst generation and 2nd generation code: surprising results
- Detailed analysis on ISP 26
- A report on the sources of discrepancies attributed to user effect has been issued (at the same time study on the computer effect)
- Writing group has been formed who will write a report on how to control the user effect.

ACTIVITIES ON THERMALHYDRAULIC APPLICATIONS

4. STATUS REPORT

- SR on the application of BE codes to licensing Distributed 1996
- The only report giving the real status of the use of B.E. codes in licensing
- The report enhances:
  - consistent approach in the countries
  - practicalities problem related to the performance of detailed uncertainty analysis

5. SIMULATORS AND PLANT ANALYZERS

- 2 specialist meetings held in Finland; the last one in Sept 97
THERMALHYDRAULIC ACTIVITIES ON ACCIDENT MANAGEMENT

◆ Establishment of a catalogue of plant states leading to core melt in PWRs and the appropriate AM measures
◆ Report issued in 1997
◆ Lot of difficulties to issue this report
◆ Possible use of this report source of information for defining the phenomena to be modelled road map for developing symptom oriented procedures and AM strategies model for developing similar efforts.

SEVERE ACCIDENT ACTIVITIES

◆ In vessel severe accident activities started around 1985
◆ First activities focused on core degradation process mainly early phase
◆ At that time difficulties to have activities on the late phase (insufficient research in member countries) (c.f. benchmark on melt relocation in lower plenum)

◆ After extensive work on early phase of core degradation, activities are moving to late phase:
  ● molten core relocation in lower plenum,
  ● Cooling of a degraded core, Quench
  ● Hydrogen production,
  ● In vessel core debris cooling through lower cavity flooding
  ● Fuel coolant interaction

◆ ACTIVITIES MAINLY RELATED TO PHENOMENOLOGICAL ASPECTS
  ● Several attempts to treat severe accident management showed the the main problem was the lack of physical knowledge.
SEVERE ACCIDENT ACTIVITIES

- ISPs
  - Core degradation
    ISP on PHEBUS SFD
    ISPs on CORA
  - Fuel coolant interaction
    ISP on FARO (ISP39)
  - Future ISPs under discussion
    PHEBUS FP
    lower head failure (PWG3)

- WORKSHOPS, SPECIALIST MEETINGS
  - Specialist meeting on FCI
  - Workshop on in vessel core debris cooling through lower cavity flooding

SEVERE ACCIDENT ACTIVITIES

- STATE OF THE ART REPORTS AND STATUS REPORT
  - Status report on quenching of a degraded core
  - Status report on molten material relocation into lower plenum
  - Other status reports in preparation (late phase degradation, hydrogen generation...)

- SEVERE ACCIDENT ACTIVITIES ON APPLICATION
  - Status report on computer effects on code results

- SEVERE ACCIDENT ACTIVITIES ON ACCIDENT MANAGEMENT
  - None
FUEL ACTIVITIES

1. HIGH BURN UP FUEL BEHAVIOR DURING RIA
   - Specialist meeting held in Cadarache in September 1995
   - Report of an ad hoc group (position paper) 1996
   - An international project based on CABRI recommended in 1996.
   - Discussion underway

2. DEGRADED FUEL BEHAVIOR DURING TRANSIENT AND ACCIDENT CONDITIONS
   - At the request of CSNI Task force on fuel safety criteria has been set up
   - Mandate: Should address fuel related safety issues in a generic sense, for new design of fuel and new fuel management
   - Final product will be a status report planned for end of 1998

CONCLUSIONS

- Brief outline of PWG2 activities has been presented
- Should mention additional interactions
  - VVER actions co-ordinated by NEA
  - CNRA activities
- PWG2 is a very efficient forum for discussion
- Very high level technical exchanges and activities
- Some activities provide reference documents
- PWG2 has created a large community of scientists working on nuclear safety
- Very deep involvement in BE codes development, assessment and now BE codes use (uncertainties evaluation studies, user effects, ....)
Best-Estimate Codes and Use of CSNI
International Standard Problems

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To be presented at the

OECD/CSNI Seminar on
"Best-Estimate Methods in Thermal-Hydraulic Safety Analysis"
Ankara, Turkey, 29 June - 1 July 1998
Best-Estimate Codes and Use of CSNI
International Standard Problems

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Abstract

Best-estimate thermalhydraulic system codes are widely used to perform safety and licensing analyses of nuclear power plants and also used in the design of advanced design reactors. Evaluation of the capabilities and the performance of these codes can be accomplished by comparing the code predictions with measured experimental data obtained on different test facilities. In this respect, parallel to other national and international programs, OECD/NEA (OECD Nuclear Energy Agency) Committee on the Safety of Nuclear Installations (CSNI) has promoted, over the last twenty two years some forty one International Standard Problems (ISPs). These ISPs were performed in different fields as in-vessel thermalhydraulic behaviour, fuel behaviour under accident conditions, fission product release and transport, core/concrete interactions, hydrogen distribution and mixing, containment thermalhydraulic behaviour. 80% of these ISPs were related to the working domain of Principal Working Group no. 2 on Coolant System Behaviour (PWG2).

The ISPs have been one of the major PWG2 activities for many years. The individual ISP comparison reports include the analysis and conclusions of the specific ISP exercises. A global review and synthesis on the contribution that ISPs have made to address nuclear reactor safety issues was initiated by CSNI-PWG2 and a short overview on this subject is given in this paper. In addition, the present status of Best-Estimate safety analysis codes is shortly summarized, since the ISP’s are used mainly for the assessment of these codes.

ISPs are providing unique material and benefits for some safety related issues. Clearly, all the technical findings and benefits provided by ISPs are still needed and contribute to advancement of nuclear safety.
1. Introduction

Large transient thermalhydraulic system codes are widely used to perform safety and licensing analyses of nuclear power plants and also used in the design of advanced design reactors. Evaluation of the capabilities and the performance of these codes can be accomplished by comparing the code predictions with measured experimental data obtained on different test facilities. In this respect, parallel to other national and international programs, OECD/NEA (OECD Nuclear Energy Agency) Committee on the Safety of Nuclear Installations (CSNI) has promoted, over the last twenty two years some forty one International Standard Problems (ISPs). The first International Standard Problem (ISP) was organized in 1975 on the famous "Edward's Blowdown Pipe" experiment. These ISPs were performed in different fields as in-vessel thermalhydraulic behaviour, fuel behaviour under accident conditions, fission product release and transport, core/concrete interactions, hydrogen distribution and mixing, containment thermalhydraulics.

A global review and synthesis on the contribution that ISPs have made to address nuclear reactor safety issues was initiated by PWG2 and writing a short overview report on this subject was approved by CSNI.

This CSNI activity on ISPs has been undoubtedly one of the most unanimous. Why such an interest and which results have been obtained are the two questions that we will try to answer in this paper.

We will provide short summary of the present status of Best-Estimate codes, some overview on the general objectives of ISPs, content and types of ISPs, and technical domains covered by ISPs. The sections dealing with technical findings and benefits to the scientific community will try to answer the two questions posed above, with some conclusions.

2. General Objectives

Within the scope of evaluating safety questions in the early 70's, it appears very soon from the technical nature of the problems that computer codes were the main tools for obtaining accident predictions from which safety measures could be verified or developed. Large programmes including code development, experimental testing and code assessment were initiated which focused in a first step on the main issue at that time, i.e. LOCA thermalhydraulics.

When OECD member countries realized the huge amount of studies which was required, they considered very rapidly that international exchanges of domestic programmes would be very beneficial for everyone. As a consequence, a period started in the 70's and in the beginning of the 80's where a large amount of results (codes and experimental data) were exchanged mostly freely between the OECD countries. CSNI contributed largely to such general exchanges, and provided its own competence in their development by initiating two kinds of particular activities:

- The first one was a direct extension of the results exchanges. It consisted in discussing and elaborating consensual conclusions on the capabilities or the deficiencies of these results in answering the safety questions towards they were directed. This generated the State of the Art Report (SOAR) activity which became one among the major CSNI activities.

- The second one was aimed to share more detailed technical experience. Recognizing that code assessment is certainly one of the key activities because it is a direct measure of the capability in predicting plant situations, it was decided to organize common exercises where a specific test was chosen for code calculations and where the code predictions from the different participants were extensively compared and discussed in reference to the experimental results. By doing such an exercise very detailed and basic technical exchanges can take place between experts. This
generated the so-called International Standard Problem (ISP) activity which is certainly a major CSNI activity and which is discussed in this paper.

3. Short Overview on the Status of Best-Estimate Safety Analysis Codes

Best Estimate LWR Analysis Codes are the major and modern tools for safety activities. They are also used extensively in the ISP activities in order to validate their application areas.

To summarize shortly the present status of Best-Estimate safety analysis codes, we will first recall for which purpose the developments of such codes were launched. Within this perspective, we will describe which products have been obtained and which scientific steps have been achieved in order to fulfil the expected initial objectives.

3.1 Initial Motivations for the Development of Best-Estimate Codes

It is well known that the decision to develop Best-Estimate (BE) codes was taken in the mid 70s based on the consideration that the calculational tools available at this date were unable to match the upcoming objectives. Indeed, in the preceding period, the lack of knowledge in the physical phenomena was circumvented by the use of the so-called conservative approach or Evaluation Model (EM) strategy. This meant that conservative assumptions were taken everywhere a phenomena was insufficiently known, in order to maximize the accident effects and as a result in order to obtain an evaluation of an upper safety bound which could not be exceeded. Even if this approach has been very useful specially for designing almost all present plants, there were two drawbacks which motivated the BE approach:

- first, it was impossible to know which was the effective safety margins specifically because of the scaling problem. Moreover this scaling problem made impossible the rigorous proof of the conservatism itself.

- second, there were several accident situations where the knowledge of the real plant response were needed like the evaluation of the Emergency Operating procedure (EOPs). An EM approach in such situations was obviously completely inadequate.

These are the two main motivations which launched the considerable research programmes on thermalhydraulics which lasted throughout the 70s and the 80s. One should remember from these motivations the fact that they were strongly related to practical applications. Consequently it is relatively to these applications that we should judge the success of these programmes and determine if there remain needs for further research.

3.2 The Achievements of Twenty Years of Safety Research on Thermalhydraulics

The twenty-year period between the mid 70s and the early 90s has been characterized by intensive research programmes including physical models development, performance of several experiments (separate effect or system experiments), improvement of numerical techniques. All the results of these researches have converged on end-products which are thermalhydraulic Best-Estimate safety codes. Some names of well known BE codes among others, ATHLET, CATHARE, CATHENA, RELAP5, TRAC-P, TRAC-B, COBRA-TRAC,...

Characteristics of some of the BE codes are summarized in table 1 [1]

If we try to characterize in some words the status of these codes, we could certainly say they are "mature" codes. The numerous applications which can be performed with these codes demonstrate undoubtedly the level of this "maturity". We can say also that they have probably almost reached the maximum of their capabilities inside the framework of the "technology" they use. To sustain this global judgement, we can consider the series of
versions which have been issued for each of these codes. Obviously considerable progress in the quality of the results and in the robustness have been made since the first versions. However, the improvements which are now added in the last versions do not modify anymore significantly the global quality. An asymptote is probably reached which demonstrates that almost all the generic capabilities of the “technology” at the basis of the development of these codes have been put to use. Finally, we can certainly declare that the issuance of these codes has allowed for the most part the objectives assigned in the early 70s, to be reached.

3.3 Scientific Steps which have been Achieved

The scientific steps which have been achieved in the present BE codes are part of the features of what we called previously the basis of the “technology” of the BE codes and which made these codes what we called in the 70s and 80s “Advanced Codes”. Let us recall briefly the most significant of these steps:

The first one concerns the two-phase flow models. Almost all BE codes now use the so called “two-fluid model” [1,2]. This model constitutes a significant step forward compared to the homogeneous equilibrium model or thermal equilibrium model with or without slip which were used before. Basically the structural progress of this model is that instead of developing global correlations in order to relate the evolution of the physical parameters of each phase (liquid and steam), this evolution is obtained as a result of the transfer laws describing at the interface the interaction between phases. This should allow a better confidence in the scale transposition as what is modelled is indeed the causes of the evolution of the parameters instead of modelling the consequences by the mean of correlations prescribing the evolution itself.

It is clear that in this approach, the elaboration of models for the transfer laws is becoming the key of the success. Considerable effort has been put on this modelling, and today each major code has a set of transfer laws or constitutive laws which cover quite extensively the accidental domain. It is probably there that we can more closely touch the asymptotic character of the improvements which are now tentatively added to the present physics of the code. The limited benefits from these additions mean first, that the objective of improving the physics relatively to the preceding generation of codes has been largely reached, and, secondly, that further improvements should certainly require a new discrete qualitative step in the basic physics [3].

All the physical modelling has been supported by an extensive experimental programme which has been simultaneously performed during this twenty-year period [4,5,6,7]. This includes basic and separate effect experiments as well as system experiments. The former was giving information for the physical modelling itself, the latter was providing the necessary data to verify the capabilities of the code in predicting situations comparable to the ones which should be encountered on a plant. Among the large and most often international programmes let us recall SEMISCALE, LOFT, PKL, LOBI which were the first programmes investigating large break accidents, BETHSY, ROSA, IV/LSTF, LOBI mod2, PKL and SPES, which were initiated after TMI in order to investigate small breaks, transients, and EOP situations, and finally the 2D-3D with UPTF.

3.4 Applications of BE codes

To get a precise picture of the overall applications of best estimate codes is somewhat difficult, since the applications are very often for specific “private” objectives e.g., commercial, safety reports, etc. and consequently they are not subject to open publication. In fact, the best picture can be obtained now by two recent documents which were elaborated by the Principal Working Group 2 from OECD/NEA and which are giving really an almost worldwide and synthesized view. The more general document is constituted of the papers of the first part of session 1 of the OECD/NEA workshop on transient thermal-hydraulic and neutronic codes requirements [8,9] held in Annapolis on November 5-8, 1996.
In this session a wide spectrum of users (France, Germany, Italy, Korea, Japan, Spain, Switzerland and United States) has given quite detailed reports on their current and anticipated use of BE thermalhydraulic codes. The second document is a status report [10] giving a summary of the utilization inside CSNI countries of best estimate methodology in safety analysis and licensing. The subject of this report is a little bit wider as it includes methodological aspects for licensing but it gives a very interesting view on how far the BE thermalhydraulic codes which are the center of the best estimate methodology are used in official procedures.

The first observation which can be made, relates to the very large number of applications which are in fact performed using best estimate thermalhydraulic codes and to the very large variety of these applications. Best estimate codes are first used in a general way for safety analyses [10] and in several cases including LOCA and transients, RIA, containment and fuel behavior. These situations are studied for operating reactors as well as for planned reactors. Some licensee calculations are performed with BE codes. But not going so far, BE codes are often considered as tools to audit the licensee calculations performed with other codes for example to verify their conservatism when they are using Evaluation Models (EM). Most analyses of operating reactor events are performed now with BE codes. One makes also use of BE codes in order to develop accident management strategies, to verify the effectiveness of backfit measures, and to analyze the emergency operating procedures. When some generic issue is raised (such as BWR stability), it is considered as logical to use BE codes. PRAs are an other domain where we find the use of BE codes. Large number of calculations for designs analyses with BE codes are related to steady state core analysis and to subchannel analysis. BE codes are going now to interfere quite often in nuclear power plant training and in simulators. Finally the domain of use of BE codes has been extended in the last years to advanced reactor designs, accidents beyond design basis accidents (DBA) and events for Eastern type reactors for which these codes were not initially designed. Such list is evidently impressive and it shows that the development of these codes has been obviously a great success.

There is a clear tendency worldwide to use in the licensing process best estimate methodology and consequently BE thermalhydraulic codes. It is obvious that the factor limiting their final appliance in practical decisions or actions and hence limiting the full benefit of the Best-Estimate, aspect, is the fact that the quality of the results is either not well qualitatively known or not well quantified. These limitations raised in fact the two main questions of, first, the determination of the range of validity of the BE code which is the objective of the code assessment and, second, the evaluation of the code uncertainties. Code assessment is evidently part of every code development, validity and applications programme. The validation matrices and CSNI International Standard Problems (ISPs) do contribute to the needed assessment process. In the next sections, the use and benefits of ISPs for the BE codes will be in detail elaborated.

4. Content and Types of ISPs

In order to better understand which contribution the ISPs have provided to safety and why this contribution has been so beneficial, it is necessary to go in some details in the specifications and the rules which were set up for the performance of ISPs.
4.1. ISP Specifications

The key feature of an ISP is that it is centered on an experimental test. This feature constitutes the main difference between ISPs and Benchmarks (following the definition made generally for these two activities in CSNI Working Groups). In a benchmark, the code predictions are made on a physical case defined theoretically for which no experimental reference can be used in the comparisons between the different calculations. On the contrary, in an ISP, the final quality of the predictions is judged on the experimental results themselves which play consequently a key role. This role has to be in fact carefully precised in the ISP specifications, and in the way the experimental data are distributed. These two points are particularly important for the ISP's significance.

- The ISP specifications include all the data necessary for performing the code calculations, for instance the geometrical characteristics, the facility instrumentation and measurements locations, the experimental boundary and initial conditions, the main actions taken during the test,... These specifications are directly derived from the test itself. In fact, during the successive ISPs, the content of the specifications has been progressively precised. The governing idea has been to put the ISP participant in the same situation as the one in which he is when he is proceeding to a plant calculation for solving a safety issue. Therefore, in the last ISPs, results like those from characterization tests giving for example pressure drops, pump characteristics, heat losses and their distribution, etc., have been added in the specifications to the test conditions itself. Furthermore, for ISPs in relation to actual plant transients, additional specifications were supplied such as the timing and degree of interventions from operators as well as plant auxiliary system conditions with their uncertainty bands.

- For the distribution of the test results, different options have been used, which lead to the distinction between different types of ISPs: open, blind, semi blind, double blind. These options due to their importance must be explained in more details.

  - Open ISP is an ISP where the participant gets all the experimental data from the very beginning. All information is open.

  - Blind ISP, on the contrary, is an ISP where the participant has no access to the test results except the test boundary and initial conditions in the general meaning discussed above. Actual test results remain locked until the calculations are made and sent to the ISP organizer. The participant is doing his calculation in a blind situation, in the same way he is doing for the calculations of plant accidents where there is no experimental data available. Blind ISPs are consequently recommended because they better reflect the real conditions in which an analyst will find himself for plant analysis. Furthermore this is reinforced by the very frequent observation that calculations are generally much better when experimental results are available at the time of calculation. This is due to the adjusting of the computer model or input data to account for the specific test results. Calculations in blind situation may avoid most of this kind of "tuning effect".

  - In reality, even in blind situations, some tuning of the code may be obtained by using in fact tests which have been already performed on the same experimental apparatus and which may be more or less similar to the one proposed for the ISP. For this reason, distinction was often made in ISPs between real blind participation cases and participations in which the participant, in fact, had opportunities to perform extensive analysis of other tests in the same experimental apparatus and consequently participation could therefore not be considered as completely blind. This is also why the concept of double blind ISP has been created.

  - A double blind ISP, is an ISP for which the participant has not access to the results of the ISP test nor to the results of any other tests performed on the experimental apparatus (except the characterization tests considered as normal additions to tests conditions). This situation represents in fact the exact real situation of the analyst when he is performing
plant calculations. Consequently double blind ISPs are especially challenging, to both the
codes and the users of the codes, and correspondingly more valuable in assessing code
performance. However such ISPs can only be performed in the very rare situations where
no test results have been already published i.e. when starting a new test facility. In this
situation, meeting the high quality of standards requested for ISP, is considerably difficult
for the laboratory running the facility as it requires a perfect control of the experimental
procedure which is in fact in contradiction with the learning phase generally experienced on
a brand new experiment. Consequently very few double blind ISPs have been practically
organised.

From the preceding discussion, it appears clearly that the more blind an ISP is, the more
significant the conclusions may be. But there are cases where the generalisation of this
statement is in fact an over-simplification.

– A semi blind ISP concept has been introduced, for example, when the prediction of the
tests involves two types of interacting phenomena and when the uncertainties on one of this
phenomena will preclude the conclusions which can be drawn on the other one. In these
specific cases it has been often decided, for preserving the ISP efficiency, that the results
concerning the first phenomena will be open and the results of the second will be blind. This
semi blind procedure has been for instance used for fuel behavior ISPs where
thermalhydraulic conditions were often open and fuel results were blind so that
uncertainties on thermalhydraulics will not prevent to draw conclusions on fuel behavior.

– As an other example where blind ISPs are not providing necessarily the most significant
conclusions, we can mention ISPs dealing with physical areas where the knowledge is at an
early stage or dealing with areas where phenomena are particularly very complex. In such
cases, blind calculations would give unusable results and valuable ISPs could only be open
ones. In those specific areas, even if results of the ISPs look encouraging, it should not be
forgotten that for such ISP, an open exercise has been chosen because of the difficulties
expected in blind calculations and that consequently, one could also expect similar
difficulties in the similar plant calculations which will be performed by the analysts in
necessarily blind-like situation.

4.2. ISP Content and Procedure

After the delivery of specifications to the participant and after the collection by the organizer
of the participant calculations, the comparisons between the different calculations and
between the calculations and the experiment can be started. A draft report comparing the
predictions to the actual data is prepared by the lead organization who collected the results.
This document is distributed to the participating organizations and discussed at a
subsequent workshop. This analysis is the most interesting part of the exercise. It provides
the basis for drawing conclusions on the capabilities and on the deficiencies of existing
calculation tools in an international framework. It is also during this analysis that the most
detailed discussions can take place between experts. It leads to a final comparison report
which is submitted to Principal Working Groups and to CSNI for issuance.

After issuing the comparison report, post-test activity on the ISP can start, if it seems useful
and technically necessary. This can be initiated immediately or after some delay (one year
or more depending on the new findings). It should involve the initial participants and
eventually new organizations. In case of blind ISP, the participation of these new
organizations will be considered as open (participation to the so called open-part of the
ISP). The initial participants have the opportunity to perform post-test sensitivity studies on
the ISP in order to better analyse the results and in order to recommend improvements to
the analytical tools. An additive to the final comparison report is generally produced. It
constitutes most often a very valuable document as it complements the first analysis which,
due to the very considerable amount of available information, may not be complete, and as
it will incorporate the new findings obtained since then.
The procedures of handling all these ISPs activities have been well established and improved when necessary (for further details see CSNI report no. 17 rev 3, November 1989, on CSNI Standard Problem Procedures, [11]) :

- For the decision about running an ISP, besides the recommendations or the wishes expressed by the Task Groups or the Principal Working Groups, the initiative is coming always and by principle from a host country which provides freely the test results and the means for making the comparisons (sometimes with collaboration of other organizations).
- After the check whether there is a consensus on the technical interest (some modification can be proposed at that stage) and that enough participation justify the organization of the ISP as offered, the specifications of the ISP are sent to the participants.
- A first workshop is held where these specifications are discussed and where all necessary information missing can be required by the participants and provided by the host country.
- A deadline for sending the contributions is agreed upon and after the receipt of the contributions, the host organization starts its comparison work. Reports of individual and collective comparisons are being written and distributed.
- A second workshop is then organized in order to collect the comments of all the participants on the draft comparison report and in order to share the different analysis. A final comparison report is then issued.
- If it is seen technically necessary and valuable, post-test activities can be initiated by the host organization. They are organised in the same way with workshops, draft reports and review process by the participants.
- During all the ISP process, special care is given for having equal treatment of all participants (diffusion of additional information, analysis of results,....).

These procedures which may look complex, are in fact one of the keys contributing to the success of the ISPs.

5. Technical Domains Covered by ISPs

A compilation of all ISPs performed between 1975 and 1997 can be found with a brief description of each ISP in the reports NEA/CSNI/R(94)19 and NEA/CSNI/R (97) 3, [12,13] and also listed in table 2.

The very first ISPs from 1975 to roughly 1980 focused on LOCA thermal-hydraulics as it was one of the main concern of that time. We find there, ISPs based on separate effect tests (Edward's blowdown pipe, CISE blowdown test, Battelle blowdown test, tube reflooding test ERSEC) and ISPs based on the two only available system experiments for PWRs at that time i.e. SEMISCALE and LOFT.

After TMI, ISPs started to move from the Large Breaks to the Small Breaks. They included ISPs on LOFT L3 small break LOCA series tests for PWRs, ROSA III test and FIX II test for BWRs. Some large break tests were still selected: PKL reflooding test, as reflooding was considered as a remaining issue; LOFT L2-5, as it was a significant "concluding" nuclear test for large breaks.

During this period (beginning 80's) two ISPs were initiated in a new domain for ISPs at that time which was the domain of thermo-mechanical fuel behavior during LOCA. These were ISPs on REBEKA test (non nuclear) and on PHEBUS LOCA test (nuclear).

In parallel to the ISPs dealing with the primary circuit, ISPs (in a first step called CASPs) were organised in the beginning of the 80's on containment experiments either system experiments (BATTENILE Model Containment) or very small scale experiment (AAEC-
Australia). These ISPs covered large break situations. They were followed in the mid 80's by ISPs on HDR containment tests (Large Break on PWR) and Marviken test (BWR).

During the second half of the 80's and during the beginning of the 90's, the ISPs related to thermalhydraulics were characterized by a full and coherent series based on the experiments which were decided and built after TMI in order to well study small break and transient situations including operator actions. They included ISPs on LOBI-mod2, SPES, ROSA IV, BETHYS for PWRs (lessons learned from these ISPs are provided in [14]), and PIPER ONE for BWRs. Besides this series, one ISP investigating the effect of non condensable gases on reflood was performed (ACHILLES), and the first and only one ISP based on real plant was organized in 1988 on the DOEL 2 steam generator tube rupture event.

End of the 80's, the interest of ISPs moved clearly to severe accident area. ISPs on core degradation were held based on CORA (non nuclear) and PHEBUS SFD (nuclear). Core concrete interaction was investigated with two ISPs (SURC4 and BETA2). Containment questions and especially hydrogen problems were the subject of two ISPs based on HDR and one ISP based on NUPEC test. Finally, ISP was also organized on FALCON to investigate Fission Product behavior with simulants.

The last recent extension of domain covered by ISPs is constituted by the move towards VVER related problems with PACTEL ISP (thermalhydraulics) and CORA VVER ISP (Core degradation).

In continuation of ISPs on thermalhydraulics and severe accident, shutdown states are investigated with an ISP on BETHYSY and steam explosions with an ISP on FARO.

This overview shows the extraordinary large range of technical domains which have been covered by ISPs. These domains reflect of course the successive changes in area of concern for safety research. This demonstrates also that the concept of ISP initiated in the thermalhydraulic area and extended to several other technical areas, is certainly very productive and useful. We will now analise what are the outcome and the benefits produced by this activity and how it may explain its success.

6. Technical Findings

The basic material of the technical findings from ISP activity is made of the several predictions obtained with several codes by several code users of a given physical experiment. From this material different cross-comparisons can be made which we will now review:

- The first class of comparisons are the comparisons between code predictions and experimental results. Such comparisons are evidently contributing to the code assessment. However some particularities to this contribution should be emphasized:
  - This assessment belongs of course to the "independent" assessment. Considering the generally very large number and very large variety of participants to ISPs, the "independent" character is certainly one of the most accentuate that we can afford. For those who are thinking that the independence of assessment is a very important feature, the results of ISPs are unique.
  - The number of code calculations in the comparison between code predictions and experimental results is certainly the largest that we can imagine on a single test. Almost no individual can do such work at least because of financial limitations. Besides this number of calculations, there are numerous differences in the physical models used in the different codes. The comparisons with experimental results are then very instructive on the effect of these models differences on the capabilities to predict the experiment. Often all codes available in OECD countries (and sometimes in the world) are represented during the ISP execution. A complete international view
is then obtained on the status of the predictive capabilities of the phenomena studied in the ISP.

- It is clear that the large amount of work produced by the participants and by the organizing country requires that no mistake should be done in the process. As a consequence, the experimental test must be first very carefully selected. Therefore it is very often one of the best and one of the most significant test of the experimental programme to which it belongs. The organization of the ISP requires also that all necessary information is transmitted to the participant in a very comprehensive way. Consequently, a very high control of test results and of documentation must be done by the organizing country. This last requirement led particularly the Working Groups to define standards for test documentation. These standards are summarized in the CSNI report no. 17 [11] and have shown to be quite general and useful, in particular as they have been used in several other areas than ISP. Finally the efforts made on the test selection, on the test control and on the test documentation provide most often a technical quality of very high level to the ISPs activities.

- The high level grade of documentation obtained by following the prescribed standards and the strict selection of the tests based on their physical and safety significance make the ISPs tests very good candidates for inclusion in validation matrices. ISPs tests may often be considered as international reference tests. Their already wide distribution and their consequent availability is also a favouring factor for such choices.

- The second class of comparisons is constituted by the comparisons between different codes. It is the common experience of analysts that understanding and analysing the code responses is a very difficult exercise. Indications are most often required in order to give directions for the analyst in its search of understanding the physical models pertinence. A first group of indications is given by the analysis of the discrepancies between calculations and experimental results which has been discussed above. A second group relates to the discrepancies between the results of different codes. This last group is often very valuable because the differences of models between the codes can be quite easily identified. Consequently the analyst can focus immediately on the concerned physical models and evaluate their relative capabilities in reference with the experimental data. By the wide variety of codes used, ISPs give good opportunities for doing extensive analysis of this kind.

- The last category of comparisons that ISPs allow, is the comparison of the results obtained with the same code by different users. The major differences between the calculations with the same code can be mainly attributed to the users of the code and this effect has been called the "User Effect". Indeed this effect is a major finding of ISPs activity. It has been discovered very early by running the very first ISPs on thermalhydraulics. The development of thermalhydraulic advanced codes was expected to decrease this effect, but the last thermalhydraulic ISPs have shown that there was still a significant "user effect" with these advanced codes. Detailed studies of this effect have been made on different ISPs and specially on ISP 26 [15]. In addition to the identification of the user effect, ISPs have contributed largely to its understanding. ISPs are really providing data which are absolutely unique on this crucial subject. Even though some suggested ways to reduce the user effect have been proposed, it remains that we are quite far from controlling it. This user effect has also appeared as a generic question and not only in the thermalhydraulics area where it has been discovered. In particular the several ISPs which have been recently performed in the severe accidents area have shown the importance of such an effect.

7. Benefits to the Scientific Community

Besides the technical findings which are shared equally by everyone, ISPs are providing general benefits which are specific, depending on the way the concerned people are
involved in the ISP activity. Three types of involvement can be distinguished: the organizers (host) of the ISP, the participants to the ISP, the research managers who decide and fund research programmes and specially the ISP activity on a national level.

7.1 Benefits to the Host Organization

- For the host organization the ISP comparisons are providing a very broad analysis of the test they have proposed and certainly a very broad and detailed analysis that they would not have in any case the possibility to have done by themselves. It is always of great value, in a large experimental programme, to have at least one of the tests very extensively analysed. By the views and recommendations that are provided, it may clearly benefit to the analysis of the other tests of the experimental programme and especially when the ISP test is one of the most significant test, which is often the case as we have indicated before.

- The large effort made for the test documentation may be very valuable to the general part of the ISP test documentation and hence to the overall experimental programme documentation. For instance, the needs expressed by the participants acting like "external customers" to get a comprehensive information, will induce very often addition in the tests documents of complementary information which was not initially foreseen. This will mostly contribute in the improvement of the documentation quality.

- Similarly questions raised by external people on the tests will often improve the general quality of the tests analysis, of the experimental findings, of the test data presentation and sometimes of the test data themselves. These are certainly indirect benefits, which are not obtained in every case, but which may be sometimes really significant.

- Having a large number of participants exercising on the same test of an experimental programme is a real opportunity for the host country to have comments and feedback from the international community on the main points of interest of his programme. Recognition on an international basis can be obtained, based on really detailed technical findings and not on more or less superficial feelings. This recognition may be also accompanied by a general consensus on how to solve the physical problems, giving the way to the host country on how eventually to improve his programme. The benefit will be obvious in that case.

If benefits cannot be denied for the host country and must not be therefore forgotten, it is clear that the host country is providing most of the effort which makes the success and the interest of the ISP. This effort has a significant cost but it is thanks to the results of this effort that other involved people are greatly benefitting from this activity.

7.2 Benefits to the Participants

- The first benefit gained by the participant besides the technical findings to which he contributes, is certainly that ISP is an opportunity for him to have a privileged access to information on the experimental programme from which the ISP test has been selected. This benefit is all the more substantial since the corresponding experimental programme is a key programme in the safety research strategy. This is often the case and particularly for experimental programmes based on large systems installations. Two cases, at this point, must be distinguished:

  - In the first one, the participant, through bilateral agreement with the country providing the ISP, has already access to the experimental programme to which the ISP belongs and sometimes even in a broader way. Nevertheless, the experience shows that very often the high quality of standard of the ISP documentation remains for him the best way to enter practically in the total programme. Benefit of access through ISP is still real and may be considerable.

  - In the second case, the participant has no bilateral agreement. This occurs specially with the small countries. For this participant, as far as tests of multilateral programmes are generally not made freely available until some years have passed,
the interest is obvious. ISP is for him a unique means to have access to detailed information and the benefit of access is crucial especially when the ISP test belongs to a major experimental programme.

Given that ISPs are chosen to cover the most important phenomena, and since an active participation of the recipient of the information is required through the ISP exercise, ISP is certainly a very efficient way for disseminating test results and for providing the subsequent important information on particular safety issue. As it has been seen before, this provides benefits especially to the small countries. But this serves also the interests of the major nuclear countries for whom it is very important to reach by means of an efficient exchange of information, a really world-wide good quality of safety related studies.

- The second benefit to the participants is that the ISPs give them an opportunity to have detailed technical discussions. These discussions are often going further than the ISP itself. Starting from exchanges of ideas on physical models related to the ISP, more basic questions are very often discussed such as questions on scaling, numerics, uncertainties, user effects. These questions which are sometimes treated "per se" in other CSNI activities, find here an excellent ground for discussion as they can be related to an experience commonly gained at the occasion of exercising on an ISP.

- Doing the ISP calculation provides to the participant a mean for evaluating his own capabilities in predicting the phenomena observed in the ISP. This is particularly true for blind ISPs. The benefits from this evaluation are strongly dependent on the participant:
  - For everyone, it represents a kind of competition which can be very stimulating and consequently positive. The idea of competition has never been really expressed explicitly in the ISP activity, but it is one of the major reasons for having strict rules in the organization of the ISP aimed to have an equal treatment of every participants as explained before.
  - For participants of major countries, the evaluation of their own capabilities in accident predictions can give a confirmation of their competence. Such confirmation has not been for most of the participants the major objective. However there have been certainly few cases where, related to this confirmation, participations to ISP have been cancelled because there was a fear that the results will not be as good as wanted for demonstrating this competence. This is clearly a drawback which cannot be avoided as far as comparisons are made between personal contributions. Most often, for this reason, the personal character was attenuated in the comparison reports by using for example cabalistic signs, but it never could be completely excluded.
  - For participants of small countries this evaluation has completely different goals and meanings. These countries are not generally developing the codes and are getting them through international agreements. For these countries, the participation in ISP is often a privileged and unique opportunity to perform by themselves independent assessment of the code that they have received and to exercise themselves as users of this code. The quality of the prediction is a mean for them to make their own judgement on the tools that they are using. By repeating this on several ISPs, they get a set of tests calculations which represents an important basis for judging the capabilities that these tools may provide to them for solving safety questions.
  - Lastly there are participants who are also starting in the code assessment activity, and for whom the ISP is in fact more or less a training exercise. As trained users are absolutely required for doing safety assessment, ISPs are providing here certainly a very valuable contribution which benefits mainly to the trainee. However we have to realize that it induces significant difficulties in the sense that the contributions of experienced and inexperienced users are mixed in the comparisons and that the meanings of the discrepancies with the experimental results in both cases are not the same. It is quite difficult to get rid of this problem but it has to be taken into
account in some way especially for some of the ISPs where there are many participants comprising necessarily some beginning users.

- Finally ISPs, as some other international activities, contribute in creating a real community of research people whose objectives are to improve their knowledge of physics in view of better safety studies. The activity of the ISPs is very propitious for this, as during a period of time, specialists from various countries are gathered in a common project which is to predict the same physical test and which is to discuss in detail their own contribution and the ones of their colleagues in order to draw conclusions on the ability in predicting accidents. This creates links between research people, a better knowledge of each other, and generally provides a privileged way to reach common understanding.

7.3 Benefits to the Research Managers

The picture given by an ISP, of a set of calculated results compared to experimental results, is certainly an objective measurement of the capabilities in the prediction of physical phenomena occurring during an accident. For the Research Managers, it should be therefore a privileged tool for answering questions such as, can the issue be considered as closed or is more research effort still needed.

It is clear that some care must be taken in this process. It is well known that results of an ISP can be presented in very different ways depending if one insists on agreements or if one insists on discrepancies. Excellent agreements on all parameters never occurred. More often there are at the same time agreements on some parameters and discrepancies on others. A well balanced judgement is then necessary. Exceptionally, there is a clear tendency for large discrepancies on every parameter. The calculation results are then obviously of low quality and one could expect clear conclusions in such case. Experience gained on ISPs have shown that these situations were in fact extremely difficult to handle as far as it appears very difficult and sometimes impossible to get recognition of the absence of positive results and especially when at the same time there were ongoing research programmes and even if one could expect that this absence was probably a temporary one.

In the case of combined situation with agreement and discrepancies, one can proceed by differences. Often the same phenomena is covered by several ISPs. One can then try to evaluate which progress has been made since the last ISP which has treated the same subject. This exercise has also shown sometimes to be very difficult as progresses in the predictions are not so obvious. This fact when it occurs should help the Research Managers in touching the real difficulty of the problems to be solved and in touching the necessary time for solving them.

For the Research Managers, besides the ISPs results, several other comparisons between calculations and experiments are available and especially those coming from domestic programmes. If ISPs by their own are certainly not sufficient to get a complete view, the "plus" they provided is that the view is much broader as it is extended to almost all the interested international community. Due to this generality, the conclusions which can be drawn should be consequently much more sound and this is certainly a very important point for defining and deciding about future research programmes.

8. Conclusions

After some adjustments at the very beginning, ISPs have become a well established activity. It extended from thermalhydraulics to several other accident domains following the main concerns of nuclear safety. Its success has been constantly increasing. The importance of the number of participants (more than 30) in some of the recent ISPs have even generated some real difficulties in handling the ISP. The outcome of these exercises are constituted of both technical findings and more general benefits to involved people.
About technical findings it has to be noticed the very wide comparisons which can be made between calculations and experiments and between calculations performed with different codes. The so called "user effect" has been demonstrated by ISPs and ISPs are providing unique material for its study. ISPs are benefiting to both the host organization and the participants. The high level of quality required for this activity is an incentive for the host organization and a way to promote its programme. For the participant, ISPs are giving him a privileged access to the results, a mean of performing code assessment, and the opportunity to have detailed discussions on several technical subjects more or less related to the ISP. This is particularly important for small countries. Research Managers may also find in ISPs, material for taking decisions about future directions for research.

In conclusion, ISPs helped in meeting several safety objectives, and namely by:

- contributing to a better understanding of postulated events and the physical phenomena involved,
- comparing and evaluating the capability of best estimate codes to predict controlled experiments and thus improving the confidence in them as assessment tools for safety issues,
- suggesting necessary improvements in the computer codes,
- providing means to assess the ability of code users, providing valuable experience and know how to be used in safety analysis of nuclear plants including licensing phases,
- providing information for quantifying safety margins in current design or licensing criteria,
- enhancing scientific discussion between computer codes developers and users in different countries.

It should be noted that ISPs have also provided reliable experimental data basis for OECD/CSNI validation matrices both for separate effects tests (SET) validation matrix \([4,7]\) and integral test facility (ITF) validation matrix \([5,6,7]\). Most of the ISPs related to thermal-hydraulics have been integrated into these validation matrices.

All the technical findings and benefits which can be expected from ISPs are still needed. Therefore ISP activity should continue and one certainly can be confident that they will contribute, as they did in the past, to the improvement of nuclear safety.

References


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<table>
<thead>
<tr>
<th>Code(s) considered</th>
<th>Fluids (beyond Steam and Water)</th>
<th>Fields/Equations</th>
<th>T/H Space Dimensions</th>
<th>Numerical Method</th>
<th>Coding Language</th>
</tr>
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<tr>
<td>TRAC-P (→TRAC)</td>
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<td>2C 2M 2E</td>
<td>1D</td>
<td>- SETS</td>
<td>FORTRAN 77 (→ FORTRAN 90)</td>
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<td>- fully implicit for 0D and 1D components</td>
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<td>- semi-impl. for 3D (Δp between phases present in num. eqn guarantees hyperbolicity)</td>
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<td>2C 2M 2E</td>
<td>1D</td>
<td>two solution schemes: semi-implicit and two-step nearly-implicit (for slow transients)</td>
<td>FORTRAN 77 (FORTRAN 90 compilable)</td>
</tr>
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<td></td>
<td></td>
<td>+ gas/steam mixture + solute concentration</td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>ATHLET</td>
<td>Boron in water Up to 5 n/c's gases in steam</td>
<td>2C 1M 2E (drift flux) (2C 2M 2E used for reflood/downcomer)</td>
<td>1D</td>
<td>fully implicit</td>
<td>FORTRAN 77</td>
</tr>
<tr>
<td></td>
<td></td>
<td>2D in downcomer</td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>TUF, CATHENA</td>
<td>Heavy water n/c gas in steam</td>
<td>T: 1C 1M 1E 1C 1M 2E (dr. flux) 2C 2M 2E</td>
<td>1D</td>
<td>T: one-step and two-step semi-impl. C: one-step semi-impl.</td>
<td>FORTRAN</td>
</tr>
<tr>
<td>HETTRAN, TRAB, ARPOS</td>
<td>H, A: n/c gas in steam A: solute in water</td>
<td>4, 5, 6 eqs See Table</td>
<td>1D</td>
<td>semi-implicit</td>
<td>FORTRAN 77 (and C)</td>
</tr>
</tbody>
</table>
Table 1 (Continued): Characteristics of some of the Best-Estimate Codes [1]

<table>
<thead>
<tr>
<th>Codes considered</th>
<th>Flow regime map</th>
<th>Interfacial Exchanges</th>
<th>Wall Exchanges</th>
<th>Virtual Mass</th>
<th>Structures: conduction, etc.</th>
</tr>
</thead>
</table>
| TRAC-P (→TRAC)   | yes, explicit for hydrodynamics | - area  
- mass transfer rate  
- friction  
- htc (int-liq, int-gas, liq-gas) | - friction to liquid  
- friction to gas | included | - heat structures connecting two hydro cells  
- 1D and 2D |
| CATHARE          | no, implicit* (smooth closure law transitions) | - friction (special case for refluxing)  
- htc (gas-int, liq-int) several regimes | - friction to liquid  
- friction to gas  
- wet and dry wall htc packages* | present in 1D 2-fluid eqs | - multi-layer  
- 2D conduction near QF  
- thermo-mechanical fuel rod model |
| RELAPS/MOD3      | yes, explicit for hydrodynamics (horizontal, vertical* and "high-mixing") | - area  
- friction*  
- int. h.t. in the bulk or near walls (from wall) | - wall friction apportioned to phases  
- htc selected by complex logic considering wall T, mass flux, void fraction, phase T's, CHF, TBHF and htc fraction | included | - multi-layer (cyl., rect., spherical)  
- two-sided  
- 2D for refluxing with fine-mesh  
- fuel rod model with chabding deformation |
| ATHLET           | various drift flux model options  
- special condensation model  
- drift flux model recognizes CCFL | - friction  
- htc selection based on wall T and void fraction  
- QF's tracked by conduction-control | included in 2D parts only | - 1D  
- 3-layer  
- two-sided |

* CATHARE: smooth evolutions of flow regime transitions are explicitly written for all closure laws (except for onset of droplet entrainment and stratification criteria)
* CATHARE: special heat transfer package for the vicinity of a quench front
* RELAPS/MOD3: using adaptation of EPR drift flux model to obtain interfacial drag in bubbly and slug flow regimes; drag coeffs used otherwise. For post-CHF: inverted annular and slug are also considered
Table 2: List of CSNI International Standard Problems (ISPs), [2]

<table>
<thead>
<tr>
<th>Number</th>
<th>Completion Date</th>
<th>Title</th>
</tr>
</thead>
<tbody>
<tr>
<td>1</td>
<td>1975</td>
<td>Standard Problem 1 - Edwards' Pipe Blowdown</td>
</tr>
<tr>
<td>2</td>
<td>1975</td>
<td>Analysis of Semicircle Blowdown Test 11, LB LOCA</td>
</tr>
<tr>
<td>3</td>
<td>1977</td>
<td>CSNI Standard Problem 3; Comparison of LOCA Analysis Codes, CISE, Blowdown</td>
</tr>
<tr>
<td>6</td>
<td>1978</td>
<td>ISP-6: Calculations Comparison Report - Determination of Water Level and Phase Separation Effects During the Initial Blowdown Phase</td>
</tr>
<tr>
<td>7</td>
<td>1979</td>
<td>Comparison Report on OECD-CSNI LOCA Standard Problem No. 7: Analysis of a Reflooding Experiment, ERSEC</td>
</tr>
<tr>
<td>8</td>
<td>1979</td>
<td>Semicircle MOD1 Test S-06-03 (LOFT Counterpart Test), LB LOCA</td>
</tr>
<tr>
<td>9</td>
<td>1981</td>
<td>LOFT Test L3-1 Preliminary Comparison report, SB LOCA</td>
</tr>
</tbody>
</table>
| 10     | 1981            | Comparison Report on OECD-CSNI LOCA Standard Problem No. 10: "Refill and Reflooding Experiment in a Simulated PWR Primary System (PKL)"
| 11     | 1984            | LOFT L3-5 and L3-6 Comparison Reports, SB LOCA                       |
| 12     | 1982            | ROSA-III 5% Small Break Test, Run 912, BWR-SB LOCA                    |
| 14     | 1985            | Behaviour of a Fuel Bundle Simulator during a Specified Heatup and Flooding Period (REBEKA Experiment) (Results of Post-Test Analyses) |
| 15     | 1983            | LOCA Experiment at FIX-II Facility, BWR                               |
| 16     | 1985            | Rupture of a Steam Line within the HDR Containment Leading to an Early Two-Phase Flow: Results of Post-Test Analyses: Final Comparison Report |
| 17     | 1984            | Marviken BWR Standard Problem                                         |
| 18     | 1987            | LOBI-MOD2 Small Break LOCA Experiment A2-81: Final Comparison Report  |
| 19     | 1987            | Behaviour of a fuel rod Bundle during a large break LOCA transient with a two-peaks temperature history (PHEBUS Experiment): Final Comparison Report |
| 20     | 1988            | Doel 2 Steam Generator Tube Rupture Event: Final Report              |
| 21     | 1989            | PIPER-ONE Experiment PO-SB-7: Simulation of Small and Intermediate Break LOCA for BWRs |
| 23     | 1989            | Rupture of a large diameter pipe in the HDR containment              |
| 24     | 1989            | SURC-4 - Core-Concrete Interaction Test                              |
| 25     | 1991            | ACHILLES - N2 injection from accumulators and faster (best estimate) reflood rates |
| 26     | 1992            | ROSA-IV LSTF-Cold-Leg Small-Break LOCA Experiment: Comparison Report  |
| 27     | 1992            | BETHSY - Small Break LOCA with Loss of HP injection                  |
| 28     | 1992            | PHEBUS SFD B9+ - Experiment on the Degradation of a PWR Type Core HDR Experiment E11.2 - Hydrogen distribution inside the HDR containment under severe accident conditions: Final Comparison Report |
| 29     | 1993            | BETA II Core-Concrete Interaction Experiment (Test V5.1): Comparison Report |
| 30     | 1993            | CORA-13 Experiment on severe Fuel Damage                             |
| 31     | 1993            | FLHT-6 Experiment, canceled                                          |
| 32     | 1994            | PACTEL-VVER-440 Natural Circulation Stepwise Coolant Inventory Reduction |

160 22.06.96
Table 2: List of CSNI International Standard Problems (ISPs)  

<table>
<thead>
<tr>
<th>Number</th>
<th>Completion Date</th>
<th>Title</th>
</tr>
</thead>
<tbody>
<tr>
<td>34</td>
<td>1994</td>
<td>Falcon Experiments FAL-ISP-1 and FAL-ISP-2, Fission product transport</td>
</tr>
<tr>
<td>35</td>
<td>1994</td>
<td>NUPEC Hydrogen Mixing and Distribution Test M-7-1: Final Comparison Report</td>
</tr>
<tr>
<td>36</td>
<td>1996</td>
<td>CORA-VVER Severe Fuel Damage Experiment (Test W2)</td>
</tr>
<tr>
<td>37</td>
<td>1996</td>
<td>VANAM M3-A Multi Compartment Aerosol Depletion Test with Hygroscopic Aerosol Material-Comparison Report</td>
</tr>
<tr>
<td>38</td>
<td>1997</td>
<td>Loss of the Residual Heat Removal System during mid-loop operation (BETHSY)</td>
</tr>
<tr>
<td>39</td>
<td>1997</td>
<td>Fuel Coolant Interaction and Quenching (FARO)</td>
</tr>
<tr>
<td>40</td>
<td>(1998)</td>
<td>STORM Test SR11 - Aerosol Deposition and Resuspension in the Primary Circuit</td>
</tr>
<tr>
<td>41</td>
<td>(1999)</td>
<td>RTF Experiment on Iodine Behaviour in Containment Under Severe Accident Conditions (provisional title)</td>
</tr>
<tr>
<td>42</td>
<td>(2001)</td>
<td>PANDA test related to passive safety systems for Advanced Light Water Reactors</td>
</tr>
<tr>
<td>43</td>
<td>(1999)</td>
<td>UMCP Boron dilution test</td>
</tr>
</tbody>
</table>

**Containment Analysis Standard Problems (CASPs)**

| CASP-3 | 1983 | Final Comparison Report for Containment Standard Problem Exercise 3 |
Overview on the CSNI Separate Effects Test and Integral Test Facility Matrices for Validation of Best-Estimate Thermal-Hydraulic Computer Codes

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   CH-5232 VilligenPSI, Switzerland

2) Gesellschaft für Anlagen- und Reaktorsicherheit (GRS) mbH,
   D-85748 Garching, Germany

To be presented at the
OECD/CSNI Seminar on
"Best Estimate Methods in Thermal-Hydraulic Safety Analysis"
Ankara, Turkey, 29 June - 1 July 1998
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D-85748 Garching, Germany

Abstract

Internationally agreed Separate Effects Test (SET) and Integral Test Facility (ITF) matrices for validation of realistic thermal hydraulic system computer codes were established. ITF development is mainly for Pressurised Water Reactors (PWRs) and Boiling Water Reactors (BWRs). These matrices were established by sub-groups of the Task Group on Thermal Hydraulic System Behaviour as requested by the OECD/NEA Committee on Safety of Nuclear Installations (CSNI) Principal Working Group 2 on Coolant System Behaviour.

Firstly, the main physical phenomena that occur during considered accidents are identified, test types are specified, and test facilities suitable for reproducing these aspects are selected. Secondly, a list of selected experiments carried out in these facilities has been set down. The criteria to achieve the objectives are outlined. In this paper some specific examples from the SET and ITF matrices will also be provided. In addition, a short summary on the status of validation matrices for Russian Pressurised Water-cooled and Water-moderated Energy Reactor (WWER) is presented.

The matrices will be a guide for code validation, will be a basis for comparisons of code predictions performed with different system codes, and will contribute to the quantification of the uncertainty range of code model predictions. In addition to this objective, the construction of such a matrix is an attempt to record information which has been generated around the world over the last 25 years, so that it is more accessible to present and future workers in that field than would otherwise be the case.
1 Introduction

For the analyses of transients and loss-of-coolant accidents (LOCAs) in Light Water Reactors (LWRs) thermal-hydraulic computer codes have been developed over the last thirty years. Starting with relative simple computer codes in the early 1970's, a continuous development of the codes has been performed with respect to a more realistic description of thermal hydraulic phenomena and a more detailed system representation.

At the beginning of the 1970's, codes for the analysis of large break LOCAs had been requested. The codes were based on the homogeneous equilibrium model, assuming equal velocities and temperatures of vapour and liquid phases. The next effort in code development were directed by the demand for the simulation of transients and small break accidents. The implementation of new models allowed for the separation of vapour and liquid by gravity. The representation of primary and secondary side with control systems and balance of plant models were extended.

In the middle of the 1970's the development of a new generation of thermal-hydraulic codes were initiated to provide analytical tools for a more realistic simulation of LWR behaviour under transient and accident conditions. Thermal and mechanical non-equilibrium phenomena have been taken into account. The effects of non-condensables and boron tracking have been considered. These codes allow the simulation of transients, the entire range of break sizes as well as beyond design basis accidents including accident management procedures with operator interventions.

Parallel to the development of the analytical tools a large variety of experimental programmes have been executed to improve the understanding of thermal-hydraulic phenomena, to study system behaviour, and to provide the required data base for code development and code validation.

A very high number of separate effects tests have been performed for the development and validation of code models. Separate effects tests investigate individual phenomena under clear boundary conditions. While in the 1970's the experiments were conducted mainly on small scale test facilities, in the 1980's more attention has been directed to scaling. For example, in 1986, the first tests at the test facility UPTF, a representation of a four loop 1300 MWe PWR with upper plenum, downcomer and the main coolant pipes in full scale reactor geometry, were performed.

The overall results of the code calculations are validated mainly by data from integral test facilities representing the primary and secondary coolant systems. While in the early 1970's the experiments were focused on large break issues, in the following, up to now, parallel to the advancement in code development, integral tests have been carried out to investigate LWR system behaviour during transients, small breaks, transients under shutdown conditions, and beyond design basis accidents. In addition to the results of integral tests LWR plant data of transients or accidents are being used to validate the predictive capability of the codes.

Construction of validation matrices is an attempt to collect together the best sets of test data for code validation and improvement from the wide range of experiments that have been carried out world-wide in the field of thermal-hydraulics. The first formulation of a validation matrix was proposed by Wolfert and Frisch from GRS [1]. This activity was taken by a CSNI sub-group to establish matrices for PWR and BWR.

In addition, to set-up validation matrices for Russian Pressurised Water-cooled and Water-moderated Energy Reactor (WWER) analyses, an international Working Group was formed on the initiative of the Federal Ministry for Research and Technology (BMFT) of the Federal Republic of Germany. A further evaluation of the WWER matrices is currently performed by a CSNI Support Group. Based on these CSNI matrices the lists of phenomena have been reviewed and adopted to the characteristics of WWER-440 and WWER-1000 systems respectively, and the lists of test facilities suitable for code assessment have been completed.
2 Definitions

Computer codes simulate the system behaviour of nuclear power plant as realistic as possible ("best estimate"). These computer codes are used to investigate

- Incidents and accidents of different scenarios and their consequences,
- the effectiveness of emergency procedures.

The process carried out by comparing code predictions with experimental measurements or measurements in a reactor plant (if available) is called validation. A code or code model is considered validated when sufficient testing has been performed to ensure an acceptable level of predictive accuracy over the range of conditions over which the code may be applied. Accuracy is a measure of the difference between measured and calculated quantities taking into account uncertainties and biases in both. Bias is a measure, usually expressed statistically, of the systematic difference between a true mean value and a predicted or measured mean. Uncertainty is a measure of the scatter in experimental or predicted data [2]. The acceptable level of accuracy is judgemental and will vary depending on the specific problem or question to be addressed by the code. The procedure for specifying, qualitatively or quantitatively, the accuracy of code predictions is also called code assessment.

The international literature often distinguishes between the terms validation and verification. A mathematical model, or the corresponding computer code, is verified when it is shown that the code behaves as intended, i.e. that it is a proper mathematical representation of the conceptual model and that the equations are correctly encoded and solved. In this context, the comparison with measured values is not part of the verification process. The term verification, however, is often used synonymously with validation and qualification [2]. Therefore, the term verification has also been used in the code validation work, including comparisons between calculations and measurements.

3 Separate Effects Test Validation Matrix

In March 1987, the OECD/NEA Committee on the Safety of Nuclear Installations (CSNI) published a document that identified a set of tests which were considered to provide the best basis for the assessment of the performance of thermohydraulic codes, "CSNI Code Validation Matrix of Thermohydraulic Codes for LWR LOCA and Transients", [3], [4], and [5]. The set of tests was chosen to include examples of all phenomena expected to occur in plant transients and LOCA analyses. Tests were selected on the basis of the quality of the data, variety of scaling and geometry, and appropriateness of the range of conditions covered. A decision was made to bias the validation matrix towards integral tests in order that code models were exercised, and interacted, in situations as similar as possible to those of interest in LWR plant. This decision was taken on the assumption that sufficient comparison with separate effects tests data would be performed, and documented, by code development, that only very limited further assessment against separate effects test data would be necessary. This last expectation has proved unrealistic; it is now recognised that continued comparison of calculations with separate effects test data is necessary to underwrite particular applications of codes, especially where a quantitative assessment of prediction accuracy is required, as well as for code model improvement.

It has been decided to develop a distinct Separate Effects Test Matrix rather than extend the original CSNI Code Validation Matrix (CCVM), which consisted almost entirely of integral tests. Only in some specific cases where integral test facility data were not available, were separate effects tests used in the CCVM. The development of the separate effects test matrix was found to require an extension of the methodology employed for the CCVM both in the scope and definition of the thermal hydraulic phenomena and in the categorisation and description of facilities.

There are several reasons for the increased importance now placed on the comparison of codes with separate effects test data. Firstly, it has been recognised that the development of individual code models often requires some iteration, and that a model, however well conceived, may need refinement as the range of applications is widened. To establish a firm need for the modification or further development
of a model it is usually necessary to compare predictions with separate effects data rather than rely on inferences from integral test comparisons.

Secondly, there is the question of uncertainties in predictions of plant behaviour. A key issue concerning the application of best estimate codes to LOCA and transient calculations is quantitative code assessment. Quantitative code assessment is intended to allow predictions of nuclear power plant behaviour to be made with a well defined uncertainty. Most schemes for achieving this quantification of uncertainty rely on assigning uncertainties to the modelling by the code of individual phenomena, for instance by the determination of reasonable ranges which key model parameters can cover and still produce results consistent with data. This interest has placed a new emphasis on separate effects tests over and above that originally envisaged for model development.

In the thermohydraulic codes, the physical processes are simulated by mechanistic models and by correlations. The prediction of particular phenomena, such as level swell or counter-current flow limitation, by a code are usually dominated by one, or perhaps a few, code models. Comparison of code predictions of basic phenomena with events observed in the relatively simple situations contrived in separate effects test facilities, often allows a better assessment of the accuracy of code models than it is possible to make with data from integral tests. This may be, for instance, because steady state rather than transient observations are possible in the separate effects tests; or because in a separate effects test facility dedicated to the study of one particular phenomenon, the measurement instrumentation can be chosen more appropriately, with less need to compromise. The more highly controlled environment of the SET is likely to lead to a more systematic evaluation of the accuracy of a model across a wide range of conditions.

A further incentive to conduct separate effects tests, in addition to those carried out in integral facilities, is the difficulty encountered in scaling predictions of phenomena from integral test facilities (which of necessity are in some sense small scale) to plant applications. Where a phenomenon is known to be highly scale dependent and difficult to model mechanistically, there is a strong case for conducting separate effects tests at full scale. In general, it is desirable to have a considerable overlap of data from different facilities; successfully predicting data from different facilities provides some confirmation that a phenomenon is well understood. The main objective in producing the SET cross reference matrix is to identify the best available sets of data for the assessment, validation and, finally, the improvement of code predictions of the individual physical phenomena. While both integral test data and SET data are appropriate for code validation and assessment, for model development and improvement there should be a strong preference for SET data.

3.1 The Methodology Developed

In the process of establishing the SET validation matrix, a methodology has been developed. This methodology helps to collect and present the data and information collected in a comprehensive and systematic manner. It is a general methodology and therefore, in principal, also applicable to the other type of validation matrices (e.g. on severe accidents). The methodology can be summarised as follows:

1. Identification of phenomena relevant to two-phase flow in relation to LOCA and thermal-hydraulic transients in light water reactors (LWRs).

2. Characterisation of phenomena, in terms of a short description of each phenomenon, its relevance to nuclear reactor safety, information on measurement ability instrumentation and data base. In addition to these points, the present state of knowledge and the predictive capability of the codes is included in the characterisation of each phenomenon.

3. Setting up a catalogue of information sheets on the experimental facilities, as a basis for the selection of the facilities and specific tests [6b].

4. Forming a separate effects test facility cross-reference matrix by the classification of the facilities in terms of the phenomena they address.

5. Identification of the relevant experimental parameter ranges in relation to each facility that addresses a phenomenon and selection of relevant facilities related to each phenomenon.
6. Establishing a matrix of experiments (the SET matrix) suitable for the developmental assessment of thermal-hydraulics transient system computer codes, by selecting individual tests from the selected facilities, relevant to each phenomenon.

3.2 Forming a SET Cross-Reference Matrix

The main objective in producing the Separate Effects Test Facility Cross Reference Matrix (SET CRM) is to identify the best available sets of data for the assessment, validation and, finally, the improvement of code predictions of the individual physical phenomena. While both integral test data and SET data are appropriate for code validation and assessment, for model development and improvement there should be a strong preference for SET data.

The thermohydraulic phenomena of interest in LWR LOCA and transients are listed in Table 1. A set of basic two-phase flow and heat transfer processes which are important for the thermohydraulic codes in the form of basic constitutive relations have been added explicitly to the list under the heading "Basic Phenomena". The scope of the SET Facility CRM has been restricted to those phenomena directly affecting the thermohydraulic behaviour in a transient or LOCA.

The resulting list of 67 thermohydraulic phenomena forms one axis of the SET Facility CRM. The second axis of the Matrix consists of the 187 facilities identified as potential sources of separate effects data. The test facilities in 12 OECD member countries are compiled (Table 2) according to the country in which they operate: Canada, Finland, France, Germany, Italy, Japan, Netherlands, Sweden, Switzerland, United Kingdom, USA, Norway. An example for SET facility CRM is shown in Table 3. The SET facility CRM tables for each country can be seen in ref.[6a]. For each test facility the phenomena addressed by the corresponding experimental research programme have been indicated in these Matrix tables, yielding the SET CRM for test facilities and thermohydraulic phenomena.

The correlation between phenomena and SET Facility is assigned to one of three levels:

- suitable for model validation, which means that a facility is designed in such a way as to simulate the phenomenon assumed to occur in a plant and is sufficiently instrumented (X);
- limited suitability for model validation: the same as above with problems due to imperfect scaling, different test fluids (e.g. Freon instead of water) or insufficient instrumentation (O);
- not suitable for model validation: obvious meaning, taking into account the two previous items (-).

This Matrix shows both the number of different phenomena covered by the experimental investigation with one test facility, and the number of different facilities in which an individual phenomenon has been investigated. The test facilities differ from each other in geometrical dimensions, geometrical configuration and operating capabilities or conditions. Therefore, the number of facilities relevant to an individual phenomenon provides some indication of the range of parameters within which a phenomenon has been investigated and experimental data generated. For instance, it is obvious from the SET CRM presented in ref.[6a] that heat transfer phenomena, especially post critical heat flux, departure from nucleate boiling/dryout and quench front propagation/re-wet, were investigated in many SET facilities.

For the systematic evaluation of the capabilities of a thermohydraulic code, appropriate experiments have to be identified which provide data over the range of conditions of interest (as far as such data is available), for each phenomenon listed, Table 4 for example.

3.3 Establishing the Separate Effects Tests (SET) Matrix

For each of the 67 phenomena, a table presents the tests which have been identified as suitable for code validation with respect to that phenomenon, from the test facilities selected. The arguments for the selection of the facilities for a given phenomenon are already identified in the previous step of the methodology.

In order to try to be practical, the number of facilities has been limited to 3 on the average, though in some special cases up to 5 are used. For heat transfer, a larger number was used, because of the large
number of parameters affecting heat transfer and its high degree of importance. The total maximum number of tests has been fixed at up to 20 per phenomenon. Here a test is considered to be a set of data points involving one key parameter variation (e.g. a flooding curve at a single pressure and tube geometry). These numbers indicate the large amount of work which is necessary to assess a code.

It must be emphasised that tests have been chosen on the basis of available information: It is not always possible to determine how satisfactory data is for code validation until it is actually used (completeness of boundary condition information; measurement accuracy, internal consistency etc.) The situation of the various experimental programmes and chosen tests varies greatly in this respect.

The tests have been selected in order to cover the experimental data range as defined, knowing that the plant range is not always covered. Particular attention has been given to the geometric scaling problem and small, medium and large scale separate effect facilities have been integrated whenever possible.

As some facilities are useful with respect to several separate effects phenomena, a cross check and a tentative harmonisation of the selected tests have been made when possible, in order to try to minimise the number of input data needed for code validation.

In this matrix the selected tests are ordered following one arbitrary chosen main parameter (for example system pressure) with, optionally, additional parameters (for example, representative diameter). This will give the user an indication of the available range of data for code validation, and the possible need for additional tests.

At the bottom of the table the main references, if identified, are given for the chosen tests. The reader is supposed to have enough information in these references to be able to compute the test. Some examples of the SET matrix for selected number of phenomena are given in Table 5. Further tables for each of the 67 phenomena are given in detail in ref.[6a].

Additional information related to the type of tests, or parameter ranges for instance are also provided in the listed references. This matrix has been published as a first attempt. It may be updated by new and additional input from the owners and by remarks from the users. Nevertheless, as it is, this separate effect test matrix covers a large number of phenomena within a large range of selected parameters. If a thermal-hydraulic code is to be used to cover a certain number of phenomena then calculation of the relevant identified tests in the matrix is considered to be a basic step toward the achievement of code qualification.

4 Integral Test Facility Validation Matrices

The validation of codes is mainly based on pre-test and post-test calculations of separate effect tests, integral system tests, and transients in commercial plants. An enormous amount of test data, available for code validation, has been accumulated. In the year 1987 the Committee on the Safety of Nuclear Installations (CSNI) of the Nuclear Energy Agency (NEA) in the Organisation for Economic Co-Operation and Development (OECD) issued a report compiled by the Task Group on the Status and Assessment of Codes for Transients and ECC [3, 4, 5]. It contains proposed validation matrices for LOCA and transients, consisting of the dominating phenomena and the available test facilities, and the selected experiments.

Since the issue of the Validation Matrix Report in 1987, new tests have been performed and an update of the validation matrix was published in the year 1996 [7]. In this report a revision and update of the matrices, including experimental facilities and identified experiments was performed. Two new matrices were included, those for „accident management for a non degraded core in PWRs“ and „transients at shutdown in PWRs“. Additional phenomena and test types were identified for these new matrices. A special chapter on counter-part tests, similar tests and International Standard Problem tests was introduced in this revision of the report. Counter-part tests and similar tests in differently scaled facilities are considered highly important for code validation. International Standard Problem experiments are carefully controlled, documented and evaluated. Therefore, these experiments are a good basis for code validation, and they were included in the tables of selected experiments. Additional work was performed to describe the content of the validation matrices, i.e. the test types, the phenomena, and most of the
selected tests. A brief description of thermal-hydraulic aspects of severe accidents was included. The thermal-hydraulic codes are being extended to the thermal-hydraulics prevailing under severe accident conditions. They cannot be considered validated at the present time. Experimental data are limited. The important phenomena for severe accident conditions, with particular emphasis on the thermal-hydraulic phenomena were summarised in the report.

4.1 Integral Test Cross Reference Matrices

To systematise the selection of tests for code validation, so called „cross reference matrices“ have been established for the first step. Based on these matrices, phenomenologically well founded sets of experiments, for which comparison of measured and calculated parameters form a basis for establishing the accuracy of test calculation results, have been defined in a second step.

In the cross reference matrices the important physical phenomena which are believed to occur during the transient or LOCA, the experimental facilities suitable for reproducing these effects, and the test types of interest are listed. The relationships

- phenomenon versus test type indicate which phenomena are occurring in which test types,
- test facility versus phenomenon indicate the suitability of the test facilities for code validation of the different phenomena, and
- test type versus test facility indicates which test types are performed in which test facilities.

For PWR facilities six individual matrices were prepared, differentiating between

- large breaks,
- small and intermediate breaks for PWR with U-tube steam generators,
- small and intermediate breaks for PWR with once-through steam generators (OTSG),
- transients,
- transients at shut-down conditions,
- accident management for a non-degraded core.

The matrix for small and intermediate breaks in PWRs with once-through steam generators have been prepared to address in particular phenomena which are unique to this reactor type.

For BWR facilities two individual matrices have been prepared, differentiating between

- loss of coolant accidents (LOCA),
- transients.

In Tables 6 to 10 cross reference matrices for PWR facilities with U-tube steam generators are shown. Among the integral system test facilities the category „PWR“ is included under „test facilities“. The analysis of accidents in actual nuclear power plants is potentially valuable with reference to scaling and simulation problems. Descriptions of phenomena and test types can be found in reference [7].

The relationship phenomenon versus test type is rated at one of three levels:

- occurring: which means that the particular phenomenon is occurring in that kind of test (plus sign in the matrix);
- partially occurring: only some aspects of the phenomenon are occurring (open circle in the matrix);
- not occurring (dash in the matrix).

The relationship test facility versus phenomenon is rated at one of three levels:

- suitable for code assessment: a facility is designed in such a way as to simulate the phenomenon assumed to occur in the plant and it is sufficiently instrumented to reveal the phenomenon (plus sign in the matrix);
limited suitability: the same as above with problems due to imperfect scaling or insufficient instrumentation (open circle in the matrix);

- not suitable: obvious meaning, taking into account the two previous items (dash in the matrix).

The relationship test type versus facility is rated at one of three levels:

- performed: the test type is useful for code assessment purposes (plus sign in the matrix);

- performed but of limited use: this kind of test has been performed in the facility, but has limited usefulness for code assessment purposes, due to poor scaling or lack of instrumentation (open circle in the matrix);

- not performed (blank).

Based on these cross reference matrices, phenomenologically well founded sets of experiments have been defined in a second step. Criteria for the selection of these tests are listed in the following Section. These selected tests form a basis for establishing the accuracy of test calculation results comparing measured and calculated values. A total number of 177 PWR and BWR-specific integral tests have been selected as potential source for thermal hydraulic code validation.

4.2 Selection of Individual Tests

A number of specific experiments were selected from those facilities which are included in the cross reference matrices described before. These selected tests versus phenomena establish the individual code validation matrices. During the selection process a number of factors were considered, including:

- Typicality of facility and experiment to expected reactor conditions,

- quality and completeness of experimental data (measurement and documentation),

- relevance to safety issues,

- test selected must clearly exhibit phenomena,

- each phenomenon should be addressed by tests of different scaling (at least one test if possible)

- high priority to International Standard Problems (ISP), counterpart and similar tests (for more explanations see [7]),

- challenge to system codes.

Where counterpart tests or similar tests were identified between two or more facilities, they were included in order to address questions relating to scaling and facility design compromises. For the accident management matrix priority was given on how realistically the test represented typical accident management procedures.

5 Cross Reference Matrices for WWER Analysis (Integral and Separate Effects Tests)

A multi-national Working Group consisting of experts from Czech Republic, Finland, France, Germany, Hungary, Russia, Slovak Republic, Poland and Ukraine has been formed on the initiative of the Federal Minister for Research and Technology (BMFT) of the Federal Republic of Germany, giving the task to GRS in close co-operation with the Nuclear Protection and Safety Institute (IPSN) of France in May 1993 to elaborate the topic "Verification Matrix for Thermalhydraulic System Codes Applied for WWER Analysis".

The topic was combined with the objective of a co-operation to formulate an internationally agreed WWER-specific validation matrix as a supplement to the existing CSNI matrix for PWRs with U-tube steam generators.
Based on the CSNI cross reference matrices the lists of phenomena have been reviewed and adopted to the characteristics of WWER-440 and WWER-1000 systems respectively, and the lists of test facilities suitable for code assessment have been completed.

The above tasks have been performed successfully by the Working Group under the leadership of GRS in close co-operation with IPSN during 1993 - 1995, and the results were published by Liesch and Réocreux [8].

The selection of tests from the large number of experiments proposed has to be continued, in order to get the ones which are the most suitable for code assessment with respect to a given phenomenon or test type. In order to support the selection, detailed explanations of the choices for the selected data have to be given.

As a consequence these activities will continue under the auspices of the OECD/NEA. Therefore, in June 1995 a new Support Group has been installed to continue with the further evaluation of the matrices, concentrating on three tasks:

- description of WWER-specific phenomena and safety relevance,
- optimization of the WWER-specific code validation matrices,
- development of criteria for the data bank storage of experimental data valid for the matrices.

6 Conclusions

A systematic study has been carried out to select experiments for thermal-hydraulic system code validation. The main experimental facilities for SETs, PWRs, BWRs and WWERs have been identified.

Matrices have been established to identify, firstly, phenomena assumed to occur in LWR plants during accident conditions and secondly, facilities and tests suitable for code validation. The matrices also permit identification of areas where further research may be justified [9, 10]. While the activities for code validation matrices for SETs, PWRs and BWRs are completed at present, the work for WWER-specific code validation matrices will be continued.

A periodic updating of the matrices will be necessary to include new relevant experimental facilities and tests (e.g. investigating boron dilution or behaviour of advanced reactors) and to include improved understanding of existing data as a result of further validation.

The first volume of the SET matrix report [6a] provides cross references between test facilities and thermal-hydraulic phenomena, and lists tests classified by phenomena. As a preliminary to the classification of facilities and test data, it was necessary to identify a sufficiently complete list of relevant phenomena for LOCA and non-LOCA transient applications of PWRs and BWRs. The majority of these phenomena are also relevant to Advanced Water Cooled Reactors and to WWERs. To this end, 67 phenomena were identified for inclusion in the SET matrix. Phenomena characterisation and the selection of facilities and tests for the SET matrix are included in volume I of the report [6a]. In all, about 1094 tests are included in the SET matrix.

To validate a code for a particular LWR plant application, it is recommended that the list of tests in the relevant matrix be viewed as the phenomenologically well founded set of experiments to be used for an adequate validation of a thermal hydraulic computer code. This set of data could serve as a basis for the estimation of code accuracy and quantification of code uncertainty.

The development and application of methods to quantify uncertainties in plant calculations is a major task for the future. This requires a determination of code uncertainties, which is based on a systematic code validation. The validation matrices are a necessary prerequisite to achieve such a systematic validation.
7 References

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"CSNI Validation Matrix for PWR and BWR Codes" Proceedings of the CSNI-Specialist Meeting on Transient Two-Phase Flow Aix-en-Provence, France, edited by M. Récourex and M.C. Rubinstein, NEA/CSNI/RL(92)/12, 1992

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a) Volume I: Phenomena Characterisation and Selection of Facilities and Tests, 
b) Volume II: Facility and Experiment Characteristics. 
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CSNI Integral Test Facility Validation Matrix for the Assessment of Thermal-Hydraulic Codes for LWR LOCA and Transients. 
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[8] Liesch K., Récourex M. 
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Common Report IPSN/GRS No. 25, July 1995


NEA/CSNI/R(96)16, November 1996
<table>
<thead>
<tr>
<th>Table 1: List of Phenomena</th>
</tr>
</thead>
<tbody>
<tr>
<td>0 BASIC PHENOMENA</td>
</tr>
<tr>
<td>1 Evaporation due to Depressurisation</td>
</tr>
<tr>
<td>2 Evaporation due to Heat Input</td>
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<tr>
<td>3 Condensation due to Pressurisation</td>
</tr>
<tr>
<td>4 Condensation due to Heat Removal</td>
</tr>
<tr>
<td>5 Interfacial Friction in Vertical Flow</td>
</tr>
<tr>
<td>6 Interfacial Friction in Horizontal Flow</td>
</tr>
<tr>
<td>7 Wall to Fluid Friction</td>
</tr>
<tr>
<td>8 Pressure Drops at Geometric Discontinuities</td>
</tr>
<tr>
<td>9 Pressure Wave Propagation</td>
</tr>
<tr>
<td>1 CRITICAL FLOW</td>
</tr>
<tr>
<td>1 Breaks</td>
</tr>
<tr>
<td>2 Valves</td>
</tr>
<tr>
<td>3 Pipes</td>
</tr>
<tr>
<td>2 PHASE SEPARATION/VERTICAL FLOW WITH AND WITHOUT MIXTURE LEVEL</td>
</tr>
<tr>
<td>1 Pipes/Plena</td>
</tr>
<tr>
<td>2 Core</td>
</tr>
<tr>
<td>3 Downcomer</td>
</tr>
<tr>
<td>3 STRATIFICATION IN HORIZONTAL FLOW</td>
</tr>
<tr>
<td>1 Pipes</td>
</tr>
<tr>
<td>4 PHASE SEPARATION AT BRANCHES</td>
</tr>
<tr>
<td>1 Branches</td>
</tr>
<tr>
<td>5 ENTRAINMENT/DEENTRAINMENT</td>
</tr>
<tr>
<td>1 Core</td>
</tr>
<tr>
<td>2 Upper Plenum</td>
</tr>
<tr>
<td>3 Downcomer</td>
</tr>
<tr>
<td>4 Steam Generator Tube</td>
</tr>
<tr>
<td>5 Steam Generator Mixing Chamber (PWR)</td>
</tr>
<tr>
<td>6 Hot Leg with ECCI (PWR)</td>
</tr>
<tr>
<td>6 LIQUID-VAPOUR MIXING WITH CONDENSATION</td>
</tr>
<tr>
<td>1 Core</td>
</tr>
<tr>
<td>2 Downcomer</td>
</tr>
<tr>
<td>3 Upper Plenum</td>
</tr>
<tr>
<td>4 Lower Plenum</td>
</tr>
<tr>
<td>5 Steam Generator Mixing Chamber (PWR)</td>
</tr>
<tr>
<td>6 ECCI in Hot and Cold Leg (PWR)</td>
</tr>
<tr>
<td>7 CONDENSATION IN STRATIFIED CONDITIONS</td>
</tr>
<tr>
<td>1 Pressuriser (PWR)</td>
</tr>
<tr>
<td>2 Steam Generator Primary Side (PWR)</td>
</tr>
<tr>
<td>3 Steam Generator Secondary Side (PWR)</td>
</tr>
<tr>
<td>4 Horizontal Pipes</td>
</tr>
<tr>
<td>8 SPRAY EFFECTS</td>
</tr>
<tr>
<td>1 Core (BWR)</td>
</tr>
<tr>
<td>2 Pressuriser (PWR)</td>
</tr>
<tr>
<td>3 Once-Through Steam Generator Secondary Side (PWR)</td>
</tr>
<tr>
<td>9 COUNTERCURRENT FLOW / COUNTERCURRENT FLOW LIMITATION</td>
</tr>
<tr>
<td>1 Upper Tie Plate</td>
</tr>
<tr>
<td>2 Channel Inlet Orifices (BWR)</td>
</tr>
<tr>
<td>3 Hot and Cold Leg</td>
</tr>
<tr>
<td>4 Steam Generator Tube (PWR)</td>
</tr>
<tr>
<td>5 Downcomer</td>
</tr>
<tr>
<td>6 Surglene (PWR)</td>
</tr>
<tr>
<td>10 GLOBAL MULTIDIMENSIONAL FLUID TEMPERATURE, VOID AND FLOW DISTRIBUTION</td>
</tr>
<tr>
<td>1 Upper Plenum</td>
</tr>
<tr>
<td>2 Core</td>
</tr>
<tr>
<td>3 Downcomer</td>
</tr>
<tr>
<td>4 Steam Generator Secondary Side</td>
</tr>
<tr>
<td>11 HEAT TRANSFER:</td>
</tr>
<tr>
<td>NATURAL OR FORCED CONVECTION</td>
</tr>
<tr>
<td>1 Core, Steam Generator, Structures</td>
</tr>
<tr>
<td>SUBCOOLED/NUCLEATE BOILING</td>
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<tr>
<td>2 Core, Steam Generator, Structures</td>
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<tr>
<td>DNB/DRYOUT</td>
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<tr>
<td>RADIATION</td>
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<tr>
<td>5 Core</td>
</tr>
<tr>
<td>CONDENSATION</td>
</tr>
<tr>
<td>6 Steam Generator, Structures</td>
</tr>
<tr>
<td>12 QUENCH FRONT PROPAGATION/REWET</td>
</tr>
<tr>
<td>1 Fuel Rods</td>
</tr>
<tr>
<td>2 Channel Walls and Water Rods (BWR)</td>
</tr>
<tr>
<td>13 LOWER PLENUM FLASHING</td>
</tr>
<tr>
<td>14 GUIDE TUBE FLASHING (BWR)</td>
</tr>
<tr>
<td>15 ONE AND TWO PHASE IMPELLER-PUMP BEHAVIOUR</td>
</tr>
<tr>
<td>16 ONE AND TWO PHASE JET-PUMP BEHAVIOUR (BWR)</td>
</tr>
<tr>
<td>17 SEPARATOR BEHAVIOUR</td>
</tr>
<tr>
<td>18 STEAM DRYER BEHAVIOUR</td>
</tr>
<tr>
<td>19 ACCUMULATOR BEHAVIOUR</td>
</tr>
<tr>
<td>20 LOOP SEAL FILLING AND CLEARANCE (PWR)</td>
</tr>
<tr>
<td>21 ECC BYPASS/DOWNCOMER PENETRATION</td>
</tr>
<tr>
<td>22 PARALLEL CHANNEL INSTABILITIES (BWR)</td>
</tr>
<tr>
<td>23 BORON MIXING AND TRANSPORT</td>
</tr>
<tr>
<td>24 NONCONDENSABLE GAS EFFECT (PWR)</td>
</tr>
<tr>
<td>25 LOWER PLENUM ENTRAINMENT</td>
</tr>
</tbody>
</table>

175
| 1 - CANADA | 5.14 FOB Blowdown, ANSALDO | 5.15 GEST-SEP, SIET |
| 1.1 Elbow Flooding Rig | 5.16 GEST-GEN (20 M W SG), SIET |
| 1.2 CWIT (CANDU reactors) | 5.17 PIPEL (Blowdown), PISA |
| 1.3 Pumps | 5.18 JF Blowdown, ENEA |
| 1.4 Header Test Facility (CANDU reactors) | 6 - JAPAN |
| 2 - FINLAND | 6.1 TPFP, JAERI |
| 2.1 REWET-I | 6.2 Air/Water Horiz. Flow Loop, JAERI |
| 2.2 REWET - II | 6.3 T-Break TF (Air/Water), JAERI |
| 2.3 | 6.4 Air/Water Rod Bundle TF, JAERI |
| 2.4 VEERA | 6.5 SG U-Tube TT, JAERI |
| 2.5 | 6.6 Single Pin Heat Transf. TF, JAERI |
| 2.6 IVO-CCFL (air/water) | 6.7 SRTF (Reflood), Toshiba |
| 2.7 IVO-Thermal Mixing | 6.8 ESTA (18 Degree Sector), Toshiba |
| 2.8 IVO-Loop Seal Facility (Air/Water) | 6.9 ESTA-KP (KWU-PWR), Toshiba |
| 3 - FRANCE | 6.10 RSTF (Refill/Reflood), Toshiba |
| 3.1 MOBY-DICK | 6.11 SHEFT (Spray Heat Transf.), Toshiba |
| 3.2 SUPER MOBY-DICK | 6.12 Guide Tube CCFL TF, Toshiba |
| 3.3 CANON and SUPER CANON (Horiz) | 6.13 Swell Level Tests, Toshiba |
| 3.4 VERTICAL CANON | 6.14 CCF, JAERI |
| 3.5 | 6.15 CCF, JAERI |
| 3.6 TAPIOCA (Vertical) | 6.16 HICOF (Hitachi Core and Fuel Tests) |
| 3.7 DADINE (Vertical Tube, Inside) | 6.17 |
| 3.8 PERICLES Rectangular | 6.18 Hot Leg CCFL Rig, JAERI |
| 3.9 PERICLES Cylindrical | 7 - NETHERLANDS |
| 3.10 PATRICIA GY 1 | 7.1 BCN Boiloff/Reflood Test (36 rods) |
| 3.11 PATRICIA GY 2 | 7.2 |
| 3.12 ERSEC Tube (Inside) | 7.3 NEPTUNUS |
| 3.13 ERSEC Rod Bundle | 8 - SWEDEN |
| 3.14 OMEGA Tube (Inside) | 8.1 GOTA BWR ECC Tests |
| 3.15 OMEGA Rod Bundle | 8.2 MARVIKEN |
| 3.16 ECTHOR Loop Seal (Air/Water) | 8.3 FRIGG/FROJA |
| 3.17 COSI | 8.4 120 bar Loop |
| 3.18 SUPER MOBY-DICK TEE | 8.5 SIV |
| 3.19 PIERO (Air/Water) | 8.6 SEPA |
| 3.20 EPOPEE | 9 - SWITZERLAND |
| 3.21 EVA | 9.1 NEPTUN-I (Boiloff) |
| 3.22 SEROPS | 9.2 NEPTUN-I and II (Reflood) |
| 3.23 BETHSY Pressuriser | 9.3 PEANUT (Reflood Inside Tube) |
| 3.24 SUPER MOBY-DICK Horizontal | 10 - UNITED KINGDOM |
| 3.25 REBECA | 10.1 ACHILLES Reflood Loop |
| 3.26 ECOTRA | 10.2 THETIS Bundle |
| 4 - GERMANY | 10.3 REFLEX Tube Reflood |
| 4.1 UPTF | 10.4 Post Dryout Int. Tube (HP, Winfrith) |
| 4.2 HDR Vessel | 10.5 TITAN/9 MW Rigs |
| 4.3 BATTELLE PWR RS 16 | 10.6 High Pressure Rig |
| 4.4 BATTELLE BWR 150396 | 10.7 Post Dryout Int. Tube (LP, Harwell) |
| 4.5 Blowdown Heat Transfer RS 37 | 10.8 Air/Water Pipeline Exp. (Large Sc.) |
| 4.6 Heat Transfer Refill/Reflood RS 36 | 10.9 Hot Leg (Air/Water, Offl, Large Sc.) |
| 4.7 Steady state DNB Exp. RS 164 | 10.10 |
| 4.8 Trans. Boil. Inst. Tube (Fremo) RS 370 | 10.11 Horiz. CCFL Rig (Air/Water, Small Sc.) |
| 4.9 Rewet RS 62/184 | 10.12 Air/Water Rigs (Small Scale) |
| 4.11 LOCA Pump Behaviour RS 92 | 10.14 Single Tube Level Swell (Harwell) |
| 4.12 Therm hyd. UP-SSR RS 373 | 10.15 Single Tube Reflood (Harwell) |
| 4.13 Pressuriser-Valve RS 290, 347, 636 | 10.16 Crossflow Two-Phase Wind Tunnel |
| 4.14 Steam/Water Disch. Flow RS 93, 397 | 10.17 Loop Seal Air/Water Rig |
| 4.15 | 10.18 Hot Leg Co and CCF Rig |
| 4.16 T-Junction Test Facility (KNK) | 10.19 Single tube Reflood (Leatherhead) |
| 5 - ITALY | 10.20 Boiler Dynamics Rig |
| 5.1 Pressuriser (Vapour Plant) ENEA | 10.21 Valve Blowdown Test Facility |
| 5.2 Pressuriser Spray, TURIN | 10.22 Single Pin Reflood |
| 5.3 Pressuriser Floating, CISE | 10.23 Multipin Cluster Rig |
| 5.4 JET-I Fuel Channel SIET | 10.24 Blowdown Rig |
| 5.5 Safety VALVE SIET | 10.25 ECCS Condensation Rig |
| 5.6 Gen 3xl (Steam Generator), SIET | 10.26 1/6th Sc, Broken Cold Leg Nozzle Rig |
| 5.7 8x8 Bundle, CISE | 10.27 1/10th Scale WPR, Refill Strath Clyde |
| 5.8 FREGENE (Steam Generator) ENEA | 10.28 RH STAR Vertical Forced Circ. Loop |
| 5.9 ARAMIS (Separator) ENEA | 10.29 RHSTAR, Forced Circ. Loop |
| 5.10 Jet Condensation, TURIN | 10.30 Vertical Flow Rigs |
| 5.11 Jet Condensation, ENEA | 10.31 High Press. Steam/Water Forced Circ. |
| 5.12 CHF, ENEA | 10.32 Low Pressure Boiling Flac (Harwell) |
| 5.13 CCF, ENEA | a : info sheet available in [4, volume 2] |
| | x : selected in the SETs matrix [1, volume 1, chapter 6] |

Table 2: List of Facilities
<table>
<thead>
<tr>
<th>11 - USA</th>
</tr>
</thead>
<tbody>
<tr>
<td>11.1 LTSF 1/6 Scale Jet Pump</td>
</tr>
<tr>
<td>11.2 Univ. California SB. LP BWR</td>
</tr>
<tr>
<td>11.3 THEF Post CHF Ins. Tube</td>
</tr>
<tr>
<td>11.4 Battelle Columbus Laboratory</td>
</tr>
<tr>
<td>11.5 Wyle Lab. Marshall Steam Station TF</td>
</tr>
<tr>
<td>11.6 Miscellaneous Sources</td>
</tr>
<tr>
<td>11.7 Univ. California SB. Vert. Tube</td>
</tr>
<tr>
<td>11.8 Univ. California B. Tube Reflood</td>
</tr>
<tr>
<td>11.9 Univ. California Berkeley</td>
</tr>
<tr>
<td>11.10 Columbia rod Bundle Blowdown HT</td>
</tr>
<tr>
<td>11.11 State Univ. New York at Buffalo</td>
</tr>
<tr>
<td>11.12 State Univ. New York at Buffalo</td>
</tr>
<tr>
<td>11.13 1/50, 1/5 + 1/3 VESSEL CREARE</td>
</tr>
<tr>
<td>11.14 1/5 DC + CL CREARE</td>
</tr>
<tr>
<td>11.15 CDN DART Bubbly Flow Nozzles</td>
</tr>
<tr>
<td>11.16 VERT TUBE PL/DART Annular CCF</td>
</tr>
<tr>
<td>11.17 TUBE + CHANNEL DART Air/Water</td>
</tr>
<tr>
<td>11.18 SNFT DART BWR Spray Nozzle</td>
</tr>
<tr>
<td>11.19 CE + MIT</td>
</tr>
<tr>
<td>11.20 1-Loop Test Fac. Westinghouse</td>
</tr>
<tr>
<td>11.21 HCNTL Univ. of Cincinnati</td>
</tr>
<tr>
<td>11.22 Heat Transf. Loop Babcock and Wilcox</td>
</tr>
<tr>
<td>11.23 FLECHT SEASET Westinghouse</td>
</tr>
<tr>
<td>11.24 Univ. California Los Angeles</td>
</tr>
<tr>
<td>11.25 SCTR Univ. California LA</td>
</tr>
<tr>
<td>11.26 Univ. California Santa Barbara</td>
</tr>
<tr>
<td>11.27 Univ. California Berkeley</td>
</tr>
<tr>
<td>11.28 HST, SSTF, VSF/GE Spray Tests</td>
</tr>
<tr>
<td>11.29 Peer Loop Natural Circulation/SRI</td>
</tr>
<tr>
<td>11.30 U-Loop SG Two-Loop Test Fac/SRI</td>
</tr>
<tr>
<td>11.31 1/5 EPRI-CREARE Mixing Facility</td>
</tr>
<tr>
<td>11.32 EPRI-SAI Thermal Mixing Test Fac.</td>
</tr>
<tr>
<td>11.33 1/2 Scale Test Facility/CREARE</td>
</tr>
<tr>
<td>11.34 EPRI/Wyle Pipe Rupture Test Fac.</td>
</tr>
<tr>
<td>11.35 TFPL/INEL Tee Critical Flow</td>
</tr>
<tr>
<td>11.36 EPRI-SAI Carryover Large Dim.</td>
</tr>
<tr>
<td>11.37 PHE/PURDUE 1/2 Scale Facility</td>
</tr>
<tr>
<td>11.38 Thermal Hydr. Test Fac/ORNEL</td>
</tr>
<tr>
<td>11.39 INEL Pump Characterisation</td>
</tr>
<tr>
<td>11.40 Semi-scale/INEL</td>
</tr>
<tr>
<td>11.41 BWR-FLECHT/GE</td>
</tr>
<tr>
<td>11.42 LEHIGH Post CHF Heat Tr. Bundle</td>
</tr>
<tr>
<td>11.43 MIT Pressuriser</td>
</tr>
<tr>
<td>11.44 LS/GE Level Swell in Blowdown</td>
</tr>
<tr>
<td>11.45 HOUSTON</td>
</tr>
<tr>
<td>11.46 Co-current Hor. Flow/Northwest</td>
</tr>
<tr>
<td>11.47 ANL Power-Void Transf. Fumc. BWR</td>
</tr>
<tr>
<td>11.48 Natural Circulation Boiling/ANL</td>
</tr>
<tr>
<td>11.49 G2 Loop/Westinghouse</td>
</tr>
<tr>
<td>11.50 Air/Water TF/B. Willamette Pump</td>
</tr>
<tr>
<td>11.51 Univ. California Berkeley</td>
</tr>
<tr>
<td>11.52 MB-2 SG Transient/Westinghouse</td>
</tr>
<tr>
<td>11.53 Strat. Condens. Flow/Northwest</td>
</tr>
<tr>
<td>11.54 Critical Flow Rig/GE</td>
</tr>
<tr>
<td>11.55 Reflux Rig/Univ. Cal. St. Barbara</td>
</tr>
<tr>
<td>11.56 LTSF Blowdown Quench/INEL</td>
</tr>
<tr>
<td>11.57 LEHIGH Post CHF Vertical Tube</td>
</tr>
</tbody>
</table>

<table>
<thead>
<tr>
<th>12 - NORWAY</th>
</tr>
</thead>
<tbody>
<tr>
<td>12.1 HALDEN Reactor, Reflood Tests</td>
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*a: info sheet available in {1, volume 2}  x: selected in the SETs matrix {1, volume 1, chapter 6}*

Table 2 (Cont.): List of Facilities
<table>
<thead>
<tr>
<th>Phenomena</th>
<th>Separate Effects Test Facilities</th>
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<tr>
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</tr>
<tr>
<td>2 limited suitability for model validation</td>
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</tr>
<tr>
<td>3 not suitable for model validation</td>
<td></td>
</tr>
<tr>
<td><strong>0 BASIC PHENOMENA</strong></td>
<td></td>
</tr>
<tr>
<td>1 Evaporation due to Depressurisation</td>
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<td>1 Core, SG, Structures</td>
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Table 3: Separate Effects Test Facility Cross Reference Matrix
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<th>No.</th>
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<th>Name</th>
<th>Keywords</th>
<th>Relevant Parameters Ranges</th>
<th>Reasons for Selection or Notes</th>
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<td>DADINE (VERTICAL TUBE INSIDE)</td>
<td>Vertical tube, Steady-state, Boil-off</td>
<td>Pressure (MPa): 0.1-0.6</td>
<td>Inlet mass flow (kg/m²s): 20-150</td>
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<td>3.12</td>
<td>a x</td>
<td>ERSEC TUBE (INSIDE)</td>
<td>Tube, reflooding</td>
<td>Pressure (MPa): 0.1-0.6</td>
<td>Inlet mass flow (kg/m²s): 10-120</td>
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<td>OMEGA TUBE (INSIDE)</td>
<td>Blowdown</td>
<td>Pressure (MPa): 16</td>
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<td>3.15</td>
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<td>OMEGA ROD BUNDLE</td>
<td>Blowdown</td>
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<td>BLOWDOWN HEAT TRANSFER RS 37</td>
<td>Blowdown Rod bundle</td>
<td>Pressure (MPa): 15-1.3</td>
<td>Inlet mass flow (kg/m²s): 3828-3300</td>
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<td>a x</td>
<td>REWET (RS 62/184)</td>
<td>Reflooding, tube, single rod</td>
<td>Pressure (MPa): 0.1-0.45</td>
<td>Inlet mass flow (kg/m²s): 2-10 cm/s</td>
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<td>GEN 3x3 (STEAM GENERATOR) ENEA</td>
<td>SG Secondary, Steady-state, transient</td>
<td>Pressure (MPa): 3.5-8</td>
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<td>8x8 BUNDLE CISE</td>
<td>BWR-6 Bundle, Steady state</td>
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<td>Core heat transfer, Boil-off, Reflooding, BWR and PWR bundle</td>
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<td>Inlet mass flow (kg/m²s): 2.5-18 cm/s</td>
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Table 4: Phenomenon No. 11.4 - Heat Transfer: POST-CHF in the Core, in the Steam Generator and at Structures (Part A)
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SELECTED TESTS

References:


Table 5: Heat Transfer: Post-CHF in the Core, in the Steam Generator and at Structures
## Matrix 1
CROSS REFERENCE MATRIX FOR LARGE BREAKS IN PWRs

- **Phenomena versus test type**
  - occurring
  - partially occurring
  - not occurring

- **Test facility versus phenomenon**
  + suitable for code assessment
  o limited suitability
  - not suitable

- **Test type versus test facility**
  + performed
  o performed but of limited use
  - not performed or planned

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<th>Phenomena</th>
<th>Blowdown</th>
<th>Rollout</th>
<th>Reflood</th>
<th>CCTF 1:25</th>
<th>LOFT 1:50</th>
<th>BETHSY 1:100</th>
<th>PKL 1:145</th>
<th>LOBI 1:712</th>
<th>SEMISCALE 1:1600</th>
<th>UPTF 1:1 (a)</th>
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<td>+</td>
<td>+</td>
<td>+</td>
<td>+</td>
<td>+</td>
<td>-</td>
<td>-</td>
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</tr>
<tr>
<td>Entrainment (Core, UP)</td>
<td>o</td>
<td>o</td>
<td>+</td>
<td>o</td>
<td>o</td>
<td>o</td>
<td>o</td>
<td>0</td>
<td>0</td>
<td>+</td>
</tr>
<tr>
<td>Deentrainment (Core, UP)</td>
<td>o</td>
<td>o</td>
<td>+</td>
<td>o</td>
<td>o</td>
<td>o</td>
<td>o</td>
<td>0</td>
<td>0</td>
<td>0</td>
</tr>
<tr>
<td>1- and 2-phase pump behaviour</td>
<td>+</td>
<td>0</td>
<td>o</td>
<td>-</td>
<td>0</td>
<td>-</td>
<td>o</td>
<td>+</td>
<td>+</td>
<td>-</td>
</tr>
<tr>
<td>Noncondensable gas effects</td>
<td>-</td>
<td>0</td>
<td>o</td>
<td>-</td>
<td>+</td>
<td>+</td>
<td>-</td>
<td>+</td>
<td>+</td>
<td>-</td>
</tr>
</tbody>
</table>

<table>
<thead>
<tr>
<th>Test Facility</th>
<th></th>
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<th></th>
<th></th>
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</thead>
<tbody>
<tr>
<td>CCTF</td>
<td>-</td>
<td>0</td>
<td>o</td>
<td>o</td>
<td>o</td>
<td></td>
<td>o</td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>LOFT</td>
<td></td>
<td>+</td>
<td>+</td>
<td>+</td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>BETHSY</td>
<td>-</td>
<td>-</td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
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</tr>
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<td>PKL</td>
<td></td>
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<td>o</td>
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<td></td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>LOBI</td>
<td></td>
<td>+</td>
<td>+</td>
<td>-</td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>SEMISCALE</td>
<td></td>
<td></td>
<td></td>
<td>+</td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>UPTF</td>
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<td></td>
<td></td>
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<td></td>
<td></td>
<td></td>
<td></td>
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<td></td>
</tr>
</tbody>
</table>

**Important test parameter**
- break location/break size
- pumps off/pumps on
- cold leg injection/combined injection

(a) UPTF integral tests

Table 6: Cross Reference Matrix for Large Breaks in PWRs
<table>
<thead>
<tr>
<th>Phenomena (c)</th>
<th>Test Facility and Volumetric Scaling</th>
</tr>
</thead>
<tbody>
<tr>
<td>Natural circulation in 1-phase flow, primary side</td>
<td>+ + + + + + + + + + + + + + + +</td>
</tr>
<tr>
<td>Natural circulation in 2-phase flow, primary side</td>
<td>+ + + + + + + + + + + + + + + +</td>
</tr>
<tr>
<td>Reflux condenser mode and CCPL</td>
<td>+ + + + + + + + + + + + + + + +</td>
</tr>
<tr>
<td>Phase separation without mixture level formation</td>
<td>+ + + + + + + + + + + + + + + +</td>
</tr>
<tr>
<td>Mixture level and entrainment in SG secondary side</td>
<td>+ + + + + + + + + + + + + + + +</td>
</tr>
<tr>
<td>Stratification in horizontal pipes</td>
<td>+ + + + + + + + + + + + + + + +</td>
</tr>
<tr>
<td>Pool formation in UPRCCPL (UCSP)</td>
<td>+ + + + + + + + + + + + + + + +</td>
</tr>
<tr>
<td>Core wide void and flow distribution</td>
<td>+ + + + + + + + + + + + + + + +</td>
</tr>
<tr>
<td>Heat transfer in covered core</td>
<td>+ + + + + + + + + + + + + + + +</td>
</tr>
<tr>
<td>Heat transfer in partly uncovered core</td>
<td>+ + + + + + + + + + + + + + + +</td>
</tr>
<tr>
<td>Heat transfer in SG primary side</td>
<td>+ + + + + + + + + + + + + + + +</td>
</tr>
<tr>
<td>Heat transfer in SG secondary side</td>
<td>+ + + + + + + + + + + + + + + +</td>
</tr>
<tr>
<td>Pressurizer thermohydraulics</td>
<td>+ + + + + + + + + + + + + + + +</td>
</tr>
<tr>
<td>Surgerline hydraulics</td>
<td>+ + + + + + + + + + + + + + + +</td>
</tr>
<tr>
<td>1- and 2-phase pump behaviour</td>
<td>+ + + + + + + + + + + + + + + +</td>
</tr>
<tr>
<td>Structural heat and heat losses (a)</td>
<td>+ + + + + + + + + + + + + + + +</td>
</tr>
<tr>
<td>Noncondensable gas effects</td>
<td>+ + + + + + + + + + + + + + + +</td>
</tr>
<tr>
<td>Boron mixing and transport</td>
<td>+ + + + + + + + + + + + + + + +</td>
</tr>
<tr>
<td>PWR, LOFT, LSTF, BETHY, PKL-III, SPECS, LOBII, SEMISCALE, UPIT, TRAM (b)</td>
<td>+ + + + + + + + + + + + + + + +</td>
</tr>
</tbody>
</table>

(a) problem for scaled test facilities
(b) UPIT integral tests
(c) for intermediate breaks phenomena included in large break reference matrix may be also important

Table 7: Cross Reference Matrix for Small and Intermediate Breaks in PWRs

182
**Matrix IV**

**CROSS REFERENCE MATRIX FOR TRANSIENTS IN PWRs**

<table>
<thead>
<tr>
<th>Phenomenon versus test type</th>
<th>Test Type</th>
<th>Test Facility and Volumetric Scaling</th>
</tr>
</thead>
<tbody>
<tr>
<td>+ occurring</td>
<td>ATWS</td>
<td>Loss of feedwater, non-ATWS</td>
</tr>
<tr>
<td>a partially occurring</td>
<td></td>
<td>Loss of heat sink, non-ATWS (c)</td>
</tr>
<tr>
<td>+ suitable for core assessment</td>
<td></td>
<td>Station blackout</td>
</tr>
<tr>
<td>o limited suitability</td>
<td></td>
<td>Steam line break</td>
</tr>
<tr>
<td>- not suitable</td>
<td></td>
<td>Pressurizer break</td>
</tr>
<tr>
<td>+ type versus test facility</td>
<td></td>
<td>Reactivity disturbance</td>
</tr>
<tr>
<td>+ performed</td>
<td></td>
<td>Overcooling</td>
</tr>
<tr>
<td>o performed but of limited use</td>
<td></td>
<td>PWR 1:1</td>
</tr>
<tr>
<td>- not performed or planned</td>
<td></td>
<td>LOFT 1:50</td>
</tr>
<tr>
<td></td>
<td></td>
<td>LSTF 1:48</td>
</tr>
<tr>
<td></td>
<td></td>
<td>BETSY 1:100</td>
</tr>
<tr>
<td></td>
<td></td>
<td>PKL-III 1:124</td>
</tr>
<tr>
<td></td>
<td></td>
<td>SPES 1:400</td>
</tr>
<tr>
<td></td>
<td></td>
<td>LOBI-II 1:712</td>
</tr>
<tr>
<td></td>
<td></td>
<td>SEMISCALE 1:1000</td>
</tr>
</tbody>
</table>

| Natural circulation in 1-phase flow | + | + | + | + | + | 0 | 0 | + | + | + | + | + | + |
| Natural circulation in 2-phase flow | + | + | + | + | + | + | 0 | - | 0 | + | + | + | + | + |
| Core thermohydraulics | + | + | + | + | 0 | 0 | + | 0 | + | + | + | + | + | + |
| Thermohydraulics on primary side of SG | + | 0 | 0 | + | + | 0 | + | + | + | + | + | + | + | + |
| Thermohydraulics on secondary side of SG | + | + | + | + | + | 0 | + | + | + | + | + | + | + | + |
| Pressurizer thermohydraulics | + | + | + | + | 0 | 0 | + | + | + | + | + | + | + | + |
| Surge line hydraulics (CCFL, choking) | + | - | - | - | - | - | - | + | - | - | - | - | - | - |
| Valve leak flow (a) | + | + | + | + | + | + | - | 0 | 0 | 0 | 0 | 0 | 0 | 0 |
| 1- and 2-phase pump behaviour | + | + | + | + | + | 0 | 0 | + | 0 | 0 | 0 | 0 | 0 | 0 |
| Thermal hydraulics-nuclear feedback | + | - | - | - | - | - | - | - | - | - | - | - | - | - |
| Structural heat and heat losses (b) | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 0 |
| Boron mixing and transport | - | - | - | - | - | - | - | - | - | - | - | - | - | - |
| Separator behaviour | 0 | - | - | - | - | - | - | - | - | - | - | - | - | - |

| PWR | - | - | - | - | - | - | - | 0 |
| LOFT | + | + | + | 0 | - | - | + |
| LSTF | - | - | - | + | + | + | + |
| BETSY | - | - | - | + | - | - | + |
| PKL-III | - | - | + | + | + | + | 0 |
| SPES | - | - | - | - | - | - | - |
| LOBI-II | + | + | + | + | + | + | - |
| SEMISCALE | - | + | + | + | + | + | + |

(a) valve flow behaviour will be strongly design-dependent, specific experimental data should be used if possible
(b) problem for scaled test facilities
(c) isolation of one or more steam generators

**Table 8: Cross Reference Matrix for Transients in PWRs**

183
### Matrix V
CROSS REFERENCE MATRIX FOR TRANSIENTS AT SHUT-DOWN CONDITIONS IN PWRs

- **Phenomenon versus test type**
  + occurring
  o partially occurring
  - not occurring

- **Test facility versus phenomenon**
  + suitable for code assessment
  o limited suitability
  - not suitable

- **Test type versus test facility**
  + performed
  o performed but of
    limited use
  - not performed or planned

<table>
<thead>
<tr>
<th>Phenomena</th>
<th>Test Type</th>
<th>Test Facility and Volumetric Scaling</th>
</tr>
</thead>
<tbody>
<tr>
<td></td>
<td>Loss of RHR with no or small opening</td>
<td>Loss of RHR with large openings</td>
</tr>
<tr>
<td>Pressurization due to boiling</td>
<td>+</td>
<td>+</td>
</tr>
<tr>
<td>Reflux condenser mode and CCFL</td>
<td>+</td>
<td>0</td>
</tr>
<tr>
<td>Asymmetric loop behaviour</td>
<td>-</td>
<td>0</td>
</tr>
<tr>
<td>Flow through openings (manways, vents)</td>
<td>-</td>
<td>+</td>
</tr>
<tr>
<td>Mixture level formation in upper plenum and hot legs</td>
<td>+</td>
<td>+</td>
</tr>
<tr>
<td>Mixture level and entrainment in the core</td>
<td>+</td>
<td>+</td>
</tr>
<tr>
<td>SG syphon draining</td>
<td>-</td>
<td>-</td>
</tr>
<tr>
<td>Asymmetry due to the presence of a dam</td>
<td>-</td>
<td>-</td>
</tr>
<tr>
<td>Stratification in horizontal pipes</td>
<td>+</td>
<td>+</td>
</tr>
<tr>
<td>Phase separation in T-junctions and effect on flow</td>
<td>-</td>
<td>+</td>
</tr>
<tr>
<td>ECC mixing and condensation</td>
<td>+</td>
<td>+</td>
</tr>
<tr>
<td>Loop seal clearing and filling</td>
<td>+</td>
<td>+</td>
</tr>
<tr>
<td>Pool formation in UP/CCFL (UCSP)</td>
<td>-</td>
<td>-</td>
</tr>
<tr>
<td>Core 3D thermalhydraulics</td>
<td>+</td>
<td>+</td>
</tr>
<tr>
<td>Heat transfer in covered core</td>
<td>+</td>
<td>+</td>
</tr>
<tr>
<td>Heat transfer in partially uncovered core</td>
<td>+</td>
<td>+</td>
</tr>
<tr>
<td>Heat transfer in SG primary side</td>
<td>+</td>
<td>+</td>
</tr>
<tr>
<td>Heat transfer in SG secondary side</td>
<td>+</td>
<td>+</td>
</tr>
<tr>
<td>Pressurizer thermalhydraulics a)</td>
<td>-</td>
<td>x</td>
</tr>
<tr>
<td>Surge line thermalhydraulics a)</td>
<td>-</td>
<td>x</td>
</tr>
<tr>
<td>Structural heat and heat losses</td>
<td>0</td>
<td>-</td>
</tr>
<tr>
<td>Non-condensible gas effects</td>
<td>+</td>
<td>+</td>
</tr>
<tr>
<td>Boron mixing and transport</td>
<td>-</td>
<td>-</td>
</tr>
<tr>
<td>Thermal hydraulic-nuclear feedback</td>
<td>-</td>
<td>-</td>
</tr>
</tbody>
</table>

**Test Facility**

<table>
<thead>
<tr>
<th>LSTF</th>
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<tbody>
<tr>
<td>BETHSY</td>
<td>-</td>
<td>+</td>
<td>+</td>
<td>-</td>
</tr>
<tr>
<td>PKL III</td>
<td>-</td>
<td>-</td>
<td>+</td>
<td>-</td>
</tr>
</tbody>
</table>

(a) x is dependent on opening location
+ pressuriser manway open
- pressuriser manway shut

Table 9: Cross Reference Matrix for Transients at Shut-Down Conditions in PWRs

184
### Matrix VI
CROSS REFERENCE MATRIX FOR ACCIDENT REFERENCE MATRIX FOR A NON-DEGRADED CORE IN PWRs

- **Phenomenon versus test type**
  - Occurring
  - Partially occurring
  - Not occurring
- **Test facility versus phenomenon**
  - Suitable for code assessment
  - Limited suitability
  - Not suitable
- **Test type versus test facility**
  - Performed
  - Performed but of limited use
  - Not performed or planned

<table>
<thead>
<tr>
<th>Phenomena</th>
<th>Test Type</th>
<th>Test Facility and Volumetric Scaling</th>
</tr>
</thead>
<tbody>
<tr>
<td>Natural circulation in 1-phase flow, primary side</td>
<td>Low pressure primary side feed and bleed</td>
<td>LOFT 1:50</td>
</tr>
<tr>
<td>Natural circulation in 2-phase flow, primary side</td>
<td>Low pressure primary side feed and bleed</td>
<td>SF-1:48</td>
</tr>
<tr>
<td>Reflux condenser mode and CCFL</td>
<td>Secondary side feed and bleed</td>
<td>BESTSY 1:100</td>
</tr>
<tr>
<td>Asymmetric loop behaviour</td>
<td>Feed and bleed</td>
<td>PKL-III 1:134</td>
</tr>
<tr>
<td>Break flow</td>
<td>Feed and bleed</td>
<td>SPE-ES 1:130</td>
</tr>
<tr>
<td>Phase separation without mixture level formation</td>
<td>Primary to secondary break with multiple failures</td>
<td>LOBI-II 1:7.12</td>
</tr>
<tr>
<td>Mixture level and entrainment in SG secondary side</td>
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<td>UPTF, TRAM 1:1</td>
</tr>
<tr>
<td>Mixture level and entrainment in the core</td>
<td></td>
<td>(a) problem for scaled test facilities</td>
</tr>
<tr>
<td>Stratification in horizontal pipe</td>
<td></td>
<td>(b) UPTF integral tests</td>
</tr>
<tr>
<td>Phase separation in T-junct. and effect on breakflow</td>
<td></td>
<td></td>
</tr>
<tr>
<td>ECC mixing and condensation</td>
<td></td>
<td></td>
</tr>
<tr>
<td>Loop seal clearing</td>
<td></td>
<td></td>
</tr>
<tr>
<td>Pool formation in UPICCFL (UCSP)</td>
<td></td>
<td></td>
</tr>
<tr>
<td>Core wide void and flow distribution</td>
<td></td>
<td></td>
</tr>
<tr>
<td>Heat transfer in covered core</td>
<td></td>
<td></td>
</tr>
<tr>
<td>Heat transfer in partly uncovered core</td>
<td></td>
<td></td>
</tr>
<tr>
<td>Heat transfer in SG primary side</td>
<td></td>
<td></td>
</tr>
<tr>
<td>Heat transfer in SG secondary side</td>
<td></td>
<td></td>
</tr>
<tr>
<td>Pressurizer thermal hydraulics</td>
<td></td>
<td></td>
</tr>
<tr>
<td>Surge line hydraulics</td>
<td></td>
<td></td>
</tr>
<tr>
<td>1- and 2-phase pump behaviour</td>
<td></td>
<td></td>
</tr>
<tr>
<td>Structural heat and heat losses (a)</td>
<td></td>
<td></td>
</tr>
<tr>
<td>Noncondensable gas effects</td>
<td></td>
<td></td>
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<tr>
<td>Accumulator behaviour</td>
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<tr>
<td>Boron mixing and transport</td>
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<td></td>
</tr>
<tr>
<td>Thermal hydraulic-nuclear feedback</td>
<td></td>
<td></td>
</tr>
<tr>
<td>Separator behaviour</td>
<td></td>
<td></td>
</tr>
</tbody>
</table>

**Table 10:** Cross Reference Matrix for Accident Management for a Non-Degraded Core in PWRs
THERMAL-HYDRAULIC ACCIDENT ANALYSIS

Jozef Mišák

INTERNATIONAL ATOMIC ENERGY AGENCY
DEPARTMENT OF NUCLEAR SAFETY
DIVISION OF NUCLEAR INSTALLATION SAFETY

OEC/CSNI Seminar on “Best Estimate Methods in Thermal- Hydraulic Safety Analysis”
Ankara, Turkey, 29 June -1 July 1998
1. INTRODUCTION

According to the internationally recognized IAEA documents, including the INSAG and NUSS documents [1, 2, 3], performing safety assessment for a Nuclear Power Plant (NPP) is one of safety requirements/principles. Safety assessment should be well documented, independently reviewed and subsequently updated in the light of significant new safety information. In the NUSS documents, a more precise terminology is also introduced. As described in the Draft IAEA Safety Guide [3], safety assessment is a broad term describing the systematic process aimed at ensuring that all relevant safety requirements are met, including the principal requirements (e.g. sufficient defence in depth, account of the operating experience and safety research), plant design requirements (e.g. equipment qualification, consideration of ageing, reliability of systems through reliability and diversity) and plant systems design requirements (e.g. specific requirements on the reactor core, reactor coolant system, containment, engineered safety features, etc).

Safety assessment includes safety analysis as its essential component, but is not limited to it. By the term safety analysis an analytical study is meant by which it is demonstrated how basic safety functions, integrity of barriers against release of radioactivity and various other acceptance criteria are fulfilled for a broad range of operating conditions, initiating events and other circumstances. Two complementary methods of safety analysis, deterministic and probabilistic, are used jointly in evaluating the NPP safety.

Deterministic analysis addresses the NPP performance under specific pre-determined operational states, postulated initiating events and accident conditions, and applies a specific set of rules and acceptance criteria. Probabilistic analysis combines the likelihood of an initiating event, all potential scenarios in the development of the event and its consequences into an estimation of risk from the NPP. Probabilistic analysis requires that many deterministic scenarios be analysed to assess the probability of certain consequences for each of the scenarios.

Deterministic analysis is typically focused on neutronic, thermal-hydraulic, radiological and structural aspects, which are often analysed with different computational tools. Thus thermal-hydraulic analysis is one part, and the most frequently used one, of overall safety analysis.

A broad variation may exist in the assumptions used in the deterministic safety analysis, from realistic/best estimate conditions to pessimistic/conservative conditions. These conditions reflect the selection of models for physical processes and the combination of these models into computer codes, as well as assumptions regarding NPP initial and boundary conditions, including performance and failures of equipment and human actions. The application of a best estimate code with realistic input data is called a best estimate analysis. More typically, various conservative analyses are used according the extent and severity of conservative assumptions adopted.

The deterministic safety analysis performed for licensing purposes is typically carried out using various conservative assumptions. Therefore, the term deterministic analysis is often used as
synonymous for conservative analysis, what is however not fully justified. The term accident analysis is often used as synonymous for deterministic safety analysis.

The methodology of safety analysis has been developed significantly over the last two decades. This fact is also reflected in the present IAEA activities, which are mainly aimed at elaborating revised Nuclear Safety Standards series (NUSS). The major part of this presentation is based on the draft documents under preparation.

An overview of some the IAEA activities in thermal-hydraulic analysis is provided in Section 2. Section 3 gives a short summary of the Draft Safety Report, under preparation as a guidance for accident analysis of commercial NPP[4]. Section 4 describes adequate consideration of the best estimate approach in safety analysis. Section 5 provides a practical example of a comparison between best estimate and conservative assumptions used in the thermal-hydraulic analysis of reactor containments and illustrates the effect of these assumptions on the results of the analysis.

2. OVERVIEW OF THE IAEA ACTIVITIES IN THERMAL-HYDRAULIC ACCIDENT ANALYSIS OF NPPs

2.1 Revision of Nuclear Safety Standards

Requirements and guidance on the scope and content of accident analysis have only been partly covered in the past by various IAEA documents, including the NUSS Code on Design [5], Code on Governmental Organization [6], Safety Guides 50-SG-D11 [7] and 50-SG-G2 [8]. Several guidelines relevant to WWER and RBMK reactors have been elaborated within the framework of the IAEA Extra-Budgetary Programme on the Safety of WWER and RBMK NPPs, namely general guidelines [9], pressurized thermal shock analysis guidelines [10], ATWS guidelines [11], containment evaluation guidelines [12], guidelines for analysis of accidents during shut-down operational modes [13], and also guidelines for analysis of accidents of RBMK reactors [14]. Comprehensive IAEA guidance on accident analysis, however, did not exist.

In the documents under development, the accident analysis is covered by the following sequence of documents:

- Draft Requirements: The Safety of Nuclear Power Plants: Design. The document requires that comprehensive safety analysis, utilizing both deterministic and probabilistic approaches be performed for each NPP.
- Draft Safety Guide: Safety Assessment and Verification. The document provides basic recommendations on deterministic safety analysis, in particular on the selection of initiating events, on objectives of the safety analysis for anticipated operational occurrences, design basis accidents and beyond design basis accidents, on methods, assumptions and acceptance criteria for the analysis.
- Draft Safety Report: Guidance for Accident Analysis of Commercial Nuclear Power Plants. The document summarizes the world practice and provides detailed suggestions on how to perform accident analysis. The contents of the guidance including its Appendices is described in the Section 3.
Although the deadline for the completion of the documents is 1999-2000, all of them are already at a very advanced stage of preparation and can be made available for trial use.

2.2 Other thermal-hydraulic activities

As a part of these activities, the IAEA is maintaining and using for various projects a number of computer codes for accident analysis (including analysis of severe accidents), and organizing relevant training and fellowship programmes. At present, the following codes are available at the IAEA:
- RELAP5/3.2
- RELAP5 3D (being modified for RBMK)
- SCDAP RELAP5/3.1
- SCDAPSIM
- MELCOR 1.8.3 and 1.8.4
- CONTAIN 1.12
- DYN 3D (WWER-1000)
- STEPAN/KOBRA (RBMK)

2.3 Activities within the framework of the IAEA Regional Technical Co-operation Projects in 1998

Within the framework of the regional technical co-operation projects, the IAEA is organizing a number of meetings, which are related to accident analysis. Such meetings are organized on the request of Member States. Examples of such meetings being organized during 1998 are as follows:

- Regional Workshop on Severe Accident Management, Ljubljana, Slovenia, 12-16 October 1998
- Regional Workshop on Format and Content of Safety Analysis Reports, Kozloduy NPP, Bulgaria, 15-19 June 1998
- Regional Workshop on Regulatory Review of Accident Analysis, Bratislava, Slovakia, 6-10 July 1998 (organised jointly with the NEA)
- Regional Workshop on Application of Plant Simulators and Analyzers for Validating EOPs and developing AM Guidelines for All Types of NPPs, Cernavoda, Romania, 30 November-
  - 4 December 1998
- Information Exchange Forum on Analytical Methods and Computational Tools for NPP Safety Assessment, Obninsk, Russia, 26-30 October 1998 (organized by US DOE, IAEA supports participation from Europe countries)
- National Workshops on Training Courses on Accident Analysis and Severe Accident Analysis in China and Pakistan hold in 1998.

3. IAEA SAFETY REPORT ON GUIDANCE FOR ACCIDENT ANALYSIS

As already stated in Section 2, as a part of the revised NUSS Series, the IAEA is developing a guidance for accident analysis of commercial NPPs [4]. The objective of the guidance is to establish a set of practical suggestions, based on the best practice world-wide for performing deterministic safety analysis. The guidance includes suggestions on the selection of initiating events, acceptance criteria, computer codes, modeling assumptions, preparation of input, presentation of results and quality assurance. At present, specific characteristics of light water reactors and CANDU reactors are covered by the Appendices to the guidance, with possible future elaboration of an Appendix for
RBMK reactors. The whole spectrum of accidents and approaches is covered, including both best estimate and conservative analysis, both design basis and beyond design basis accidents, both new plants and existing plants. The guidance is at an advanced stage of preparation and in the near future will be distributed to a number of organizations for their comments.

The more detailed description of the contents of the guidance is provided below.

INTRODUCTION
- the interrelation among terms such as defence in depth, safety analysis, deterministic analysis is introduced
- the objective of the guidance is described
- scope of the guidance is specified- to be used by developing countries, both for new and existing NPPs, covering all neutronic, thermal-hydraulic, radiological and, partially, structural analysis

CLASSIFICATION OF INITIATING EVENTS
- the interrelation between key safety functions and initiating events is provided
- the grouping of initiating events is described: a) by effect of potential degradation of key safety functions, b) by cause of the initiating event, c) by frequency of the event occurrence, d) by relation of an event to the original plant design
- the concept of a bounding accident scenario is introduced

SAFETY REQUIREMENTS AND ACCEPTANCE CRITERIA
- safety requirements, as more detailed conditions derived from the need to maintain key safety functions are introduced
- global and detailed acceptance criteria as a direct measure for fulfilling of safety requirements to be proved by analysis are described
- the interrelation between acceptance criteria and probability of occurrence of a scenario is proposed

ANALYSIS OF INDIVIDUAL EVENTS
- a summary of principles is given for two basic approaches: conservative and best estimate, including uncertainties and sensitivities
- basic rules for the selection of initial conditions, availability of systems and components and operator actions for conservative analysis are presented
- the interrelation between uncertainties, sensitivities and how to determine them is discussed
- the suggestion is made to apply only best estimate codes for analysis and to use them either for full best estimate analysis with specification of uncertainties or to combine them with conservative input data

APPLICABILITY OF RESULTS
- the characteristics of the application of accident analysis for design, licensing, Emergency Operating Procedures (EOP), simulators, Probabilistic Safety Assessment (PSA), accident management and emergency planning are given
- a conservative approach is suggested for design and design modifications (including extra margins) and licensing
- best estimate analysis is suggested for preparatory and validation analysis of EOP, for validation and for defining the limits of applicability of the plant simulators, for PSA analysis, and, intentionally, as a support for accident management and emergency planning; conservative
models are, however still used to overcome the lack of information regarding the molten core behaviour.

COMPUTER CODES
- six categories of computer codes are recognized: a) reactor physics; b) fuel behaviour, c) system thermal-hydraulics, d) containment analysis including radioactivity transport features, e) atmospheric dispersion and dose analysis, f) structural analysis.
- well-known codes for each category are listed, with special attention to codes for severe accident analysis and for simulation purposes
- key phenomena to be modelled by the codes are described
- requirements on the complete set of documentation for each code are specified
- requirements and procedures for code verification and validation are described
- use of internationally recognized best estimate codes with broad range of applicability is suggested where possible
- detailed simulation codes are suggested for development and validation of EOP, accident management guidelines and training
- need for the correct use of the codes within the range of their applicability is underlined

USER EFFECT ON THE ANALYSIS
- the importance of the user qualification for the quality of the analysis is underlined and ways of reducing the user's negative effect are described
- suggestions on how a qualified user can be trained are given
- suggestions for code developers and users on the reduction of user effects are provided, including systematic quality assurance, improved user guidelines, continued code improvements, independent code validation by each organization, systematic user training, participation in software user groups.

INPUT DATA PREPARATION
- basic steps for input data preparation and documentation, including collection of data from reliable sources, creation of an engineering handbook and input decks, and checking the quality of data are described
- input data verification and validation are stated as a necessary requirement and practical ways for performing these are suggested, including checking the adequacy of nodalization, energy and mass balance in the modelled system, checking the behaviour and response of individual components, checking the steady state conditions for various plant regimes, comparison with start-up tests and operational transients

PRESENTATION AND EVALUATION OF RESULTS
- suggestions are provided on format and structure of reports, including a list of parameters to be presented
- suggestions are made on detail review of results prior to their utilization

SUGGESTIONS ON QUALITY ASSURANCE IN ACCIDENT ANALYSIS
- relevant basic principles of QA are presented, including development of formalized procedures, specification of responsibilities, proper recording and documentation, independent reviewing, effective control of non-conformance and corrective actions, adequate archiving.
APPENDIX A. SPECIFIC INFORMATION RELATED TO LIGHT WATER REACTORS (LWR)

1. INTRODUCTION
2. INITIATING EVENTS AND THEIR CATEGORIZATION
3. ACCEPTANCE CRITERIA
4. SELECTION OF INITIAL CONDITIONS
5. ANALYSIS OF CORE COOLING AND SYSTEM PRESSURIZATION FOR REACTIVITY ACCIDENTS
   5.1. Control rod ejection
   5.2. Control rod withdrawal
   5.3. Control rod maloperation
   5.4. Incorrect connection of an inactive RCS loop
   5.5. Boron dilution
   5.6. Inadvertent loading of a fuel assembly in an improper position
6. ANALYSIS OF CORE COOLING AND SYSTEM PRESSURIZATION FOR DECREASE OF REACTOR COOLANT FLOW
   6.1. Single or multiple MCP trips
   6.2. Inadvertent closure of a MIV in a RCS loop
   6.3. Seizure of one MCP or shaft break for one MCP
7. ANALYSIS OF SYSTEM PRESSURIZATION FOR INCREASE OF REACTOR COOLANT INVENTORY
   7.1. Inadvertent actuation of ECCS
   7.2. Malfunction of chemical and volume control system
8. ANALYSIS OF CORE COOLING AND SYSTEM PRESSURIZATION FOR INCREASE OF HEAT REMOVAL BY THE SECONDARY SIDE
   8.1. Steam line breaks
   8.2. Inadvertent opening of steam releasing valves
   8.3. Secondary pressure control malfunction with increase of steam flow rate
   8.4. Feedwater system malfunction
9. ANALYSIS OF CORE COOLING AND SYSTEM PRESSURIZATION FOR DECREASE OF HEAT REMOVAL BY THE SECONDARY SIDE
   9.1. Feedwater line break
   9.2. Feedwater pump trips
   9.3. Reduction of the steam flow from the steam generators due to various reasons
10. ANALYSIS OF CORE COOLING FOR LOCA
11. ANALYSIS OF CONTAINMENT BY-PASS DUE TO LEAKS FROM THE PRIMARY TO THE SECONDARY SYSTEM
12. ANALYSIS OF ATWS
13. ANALYSIS OF BORON DILUTION ACCIDENTS
14. ANALYSIS OF PTS
15. ANALYSIS OF RHR DEGRADATION DURING SHUTDOWN OPERATIONAL MODES
16. ANALYSIS OF PRESSURE-TEMPERATURE TRANSIENTS IN THE CONTAINMENT
   16.1. Short-term containment pressurization
   16.2. Long-term pressure-temperature transients in the containment
17. ANALYSIS OF RADIOACTIVITY TRANSPORT DURING DBA
18. ANALYSIS OF SEVERE ACCIDENTS
APPENDIX B. SPECIFIC INFORMATION RELATED TO PRESSURIZED HEAVY WATER REACTORS (PHWR)- CANDU

1. INTRODUCTION
2. SELECTION OF INITIATING EVENTS
3. CATEGORIZATION OF INITIATING EVENTS
4. HIGH-LEVEL ACCEPTANCE CRITERIA
5. MAJOR COMPUTER ANALYSIS TOOLS REQUIRED FOR DBAs
6. SELECTION OF INITIAL CONDITIONS
7. TYPICAL INITIATING EVENTS
   7.1. Large Heat Transport System LOCA
   7.2. Small Heat Transport System LOCA
   7.3. Single Channel Events
   7.4. Single Steam Generator Tube Rupture
   7.5. Multiple Steam Generator Tube Failure
   7.6. Loss Of Forced Circulation
   7.7. Loss Of Reactivity Control (LORC)
   7.8. Loss Of Pressure And Inventory Control (Primary)
   7.9. Main Steam Line Breaks
   7.10. Feedwater System Failures
   7.11. Loss of Secondary Side Pressure Control
   7.12. Loss of Shutdown Heat Sink
   7.13. Moderator System Failures
   7.14. Shield Cooling System Failures
   7.15. Severe Accidents
8. UNCERTAINTY ANALYSIS
9. REPORTING OF RESULTS
10. REFERENCES
4. ROLE OF THE BEST ESTIMATE ANALYSIS

In the new Draft INSAG-3 document[2], as well as in the IAEA Safety Standard Series documents [1, 3], the use of conservative rules and criteria incorporating safety margins is required in NPP design. INSAG-10 on Defence in Depth in Nuclear Safety [15] characterizes appropriate conservatism as one out of three prerequisites (together with QA and safety culture) for an effective implementation of the defence in depth concept. INSAG-10 also requires that in the deterministic analysis of postulated events, conservative assumptions are made in all steps of the analysis to show that the response of the plant and its safety systems allow the safety targets to be met with adequate margins. INSAG-10 also indicates, consistently with the IAEA Draft Safety Guide [3], how to apply the requirement of conservatism in analysis related to individual levels of defence in depth. The appropriate conservatism, as required by all above mentioned documents, is also interpreted by the IAEA document IAEA-TECDOC-986, devoted to the next generation of light water reactors [16]. According to this document, conservatism should be fully applied at the first three levels of defence, with the most evident degree, rigour and formality at Level 3. At Level 4 and 5, best estimate considerations are increasingly important and recommended, and only when this is not possible, reasonably conservative assumptions should be made which take into account the uncertainties in the understanding of the physical processes. Conservative assumptions should be applied at the first three levels of defence in all phases of the design, and also in the review of modifications, assessment of ageing effects, periodic safety reassessment as well as in regulatory review and subsequent licensing decisions. Best estimate analysis should be applied in the elaboration of accident management measures and emergency plans, and also in the validation of EOP, plant simulators and in all PSA related analysis.

It is understood [16] that with the advances in knowledge through research, testing, improved computer models, increased operating experience, etc., many uncertainties in the plant behaviour are being removed. This enables a reduction in the use of intentionally conservative or bounding assumptions in accident analysis.

There is already a broad consensus that, in view of the current maturity of best estimate computer codes, best estimate codes should be used even for conservative accident analysis, while traditional conservative codes with over-conservative and often unrealistic descriptions of physical phenomena should be abandoned.

In accordance with the IAEA documents [3, 4], there are two acceptable ways offered to achieve adequate conservatism:
- Use of the best estimate code with a reasonably conservative selection of input data, which should ensure that the actual plant response related to a selected criterion is bounded by the conservative value for that response
- Use of the best estimate code with realistic assumptions on initial and boundary conditions; this approach ensures that the predicted plant behaviour with given uncertainty includes the actual value. Statistically combined uncertainties for the plant conditions and code models are therefore a necessary part of the analysis.

A best estimate approach with the uncertainty estimates provides a direct measure of safety margins, which can later on be eliminated by more advanced investigations. On the other hand, the use of the full best estimate approach is not typically possible or desirable, because of the difficulty of quantifying code uncertainties for every phenomena and every accident sequence. In particular, the
lack of experimental data for portions of severe accidents precludes the complete definition of code uncertainties for these accidents. Therefore, the combination of the best estimate code and conservative input is particularly attractive at present.

5. COMPARISON OF THE REALISTIC AND THE CONSERVATIVE APPROACH FOR THERMOHYDRAULIC ANALYSIS OF LARGE DRY CONTAINMENT

The analysis of pressure-temperature transients in reactor containments is also one of the essential parts of thermal-hydraulic accident analysis. A typical set of acceptance criteria for the evaluation of acceptability for this kind of safety analysis is as follows [4]:

1. The calculated peak containment pressure should be lower than the containment design pressure and calculated minimum containment pressure should be higher than the corresponding acceptable value.
2. Differential pressures acting on the containment internal structures important for the containment integrity should be maintained below acceptable values.

Other relevant criteria typically require limitation of doses/releases into the environment, prevention of the containment failure due to pressure/temperature loads following severe accidents, etc.

Table 1. shows how conservative assumptions have been typically adopted in accident analysis. The influence of some of these assumptions is also shown, as roughly estimated for a pressure peak in a reference full pressure containment of a WWER 1000 unit following a double ended full size break of the main circulation line. The reference containment has a total internal volume of 52,500 m³, of that 6,800 m³ are the volume of the steam generators compartment interconnected through an opening of 62.6 m² with the main containment volume. The reference peak containment pressure was 471.7 kPa, the reference pressure difference between two compartments was 68 kPa.
Table 1. COMPARISON OF REALISTIC AND CONSERVATIVE ASSUMPTIONS FOR CONTAINMENT THERMOHYDRAULIC ANALYSIS (REFERENCE LARGE DRY CONTAINMENT OF WWER - 1000 UNIT)

**CRITERION:** Maximum containment pressure and pressure difference between compartments

<table>
<thead>
<tr>
<th>PARAMETER</th>
<th>REALISTIC ASSUMPTION</th>
<th>CONSERVATIVE ASSUMPTION</th>
<th>INFLUENCE ON PEAK PRESSURE FOLLOWING MAXIMUM LOCA</th>
</tr>
</thead>
<tbody>
<tr>
<td><strong>SOURCES OF ENERGY</strong></td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>Volume of the primary coolant</td>
<td>Design value of the free volume</td>
<td>Design value increased by ~ 2%</td>
<td>+ 1.1%</td>
</tr>
<tr>
<td>Reactor power and coolant parameters</td>
<td>100% power, nominal parameters</td>
<td>Continuous operation at 102% power, parameters at their upper range</td>
<td></td>
</tr>
<tr>
<td>Break size</td>
<td>Break size limited at the crack arrest size</td>
<td>Double-ended full size break of the main circulation line</td>
<td></td>
</tr>
<tr>
<td>Break opening time</td>
<td>Break developing in time and probably monitored by LBB systems</td>
<td>Instantaneous full size break</td>
<td></td>
</tr>
<tr>
<td>Blow-down mass flow rate</td>
<td>Best-estimate fluid outflow correlations</td>
<td>Conservative fluid outflow correlations</td>
<td></td>
</tr>
<tr>
<td>Residual heat of the reactor</td>
<td>Best estimate prediction in accordance with power history</td>
<td>Prediction for continuous 102% power, conservatively increased by 20%</td>
<td></td>
</tr>
<tr>
<td>Timing of thermal power from various sources of energy</td>
<td>Thermal power variation in time, partially compensated by heat removal</td>
<td>Summing up of energy accumulated in coolant with integral of energy from other sources until the time of peak</td>
<td></td>
</tr>
<tr>
<td><strong>INTERNAL THERMOHYDRAULICS OF CONTAINMENT ATMOSPHERE</strong></td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>Containment free volume</td>
<td>Design value</td>
<td>Design value reduced by ~ 3%</td>
<td>+ 0.7%</td>
</tr>
<tr>
<td>Containment initial parameters (pressure, temperature, humidity)</td>
<td>Nominal parameters</td>
<td>Conservative values (for full pressure containment - high pressure, low temperature, low humidity)</td>
<td>+ 1%</td>
</tr>
<tr>
<td>Containment leak rate</td>
<td>Leak rate in accordance with measurements</td>
<td>Not considered</td>
<td></td>
</tr>
<tr>
<td>Heat absorption in walls and structures</td>
<td>Intensive during the blow-down, significant later on</td>
<td>Not considered</td>
<td>+ 3.3%</td>
</tr>
<tr>
<td>Thermodynamic non-equilibrium between liquid and gaseous phases</td>
<td>Certain non-equilibrium based on geometry, composition of atmosphere, circulation</td>
<td>Full thermodynamic non-equilibrium</td>
<td>+ 6% comparing to full equilibrium</td>
</tr>
<tr>
<td>PARAMETER</td>
<td>REALISTIC ASSUMPTION</td>
<td>CONSERVATIVE ASSUMPTION</td>
<td>INFLUENCE ON PEAK PRESSURE FOLLOWING MAXIMUM LOCA</td>
</tr>
<tr>
<td>----------------------------------------------------</td>
<td>----------------------------------------------------------</td>
<td>---------------------------------------------------------</td>
<td>--------------------------------------------------</td>
</tr>
<tr>
<td>Sub-division of the containment into compartments</td>
<td>Sub-division considered</td>
<td>One single volume considered</td>
<td>+ 2 %</td>
</tr>
<tr>
<td>Water carry-over between compartments</td>
<td>Typically 10-30%, time dependent, influenced by many factors</td>
<td>Homogenous mixture of steam and water considered</td>
<td>+ 0.9-+1.8 % on peak pressure, + 29-44 % on pressure differ.</td>
</tr>
<tr>
<td>Inter-compartment flow model</td>
<td>Realistic flow pattern</td>
<td>&quot;Frozen&quot; flow model</td>
<td></td>
</tr>
<tr>
<td>Cooling effect of ECCS coolant escaping from RCS before entering containment sump</td>
<td>Certain condensation of steam on water stream</td>
<td>Not considered</td>
<td></td>
</tr>
<tr>
<td>Internal circulation flow</td>
<td>Existing, enhancing heat transfer to walls and structures</td>
<td>Not considered</td>
<td></td>
</tr>
<tr>
<td>Flow path hydraulic resistance</td>
<td>Realistic resistance estimation</td>
<td>Flow through the opening</td>
<td>~ 0 % on peak pressure, + 13.7 % on pressure difference</td>
</tr>
<tr>
<td>CONTAINMENT SYSTEMS AND OTHER SAFETY SYSTEMS</td>
<td></td>
<td></td>
<td>Items important for long-term effects and smaller LOCA only</td>
</tr>
<tr>
<td>Availability of external power supply to systems</td>
<td>Power supply continuously available</td>
<td>Loss of power supply, delay due to start of DG</td>
<td></td>
</tr>
<tr>
<td>Emergency core cooling system</td>
<td>All trains operating, removing residual heat</td>
<td>1 train operating, residual heat partly increasing containment atmosphere</td>
<td></td>
</tr>
<tr>
<td>Vent and fan cooler systems</td>
<td>All systems in operation at nominal capacity</td>
<td>1 system in operation, at minimum estimated capacity</td>
<td></td>
</tr>
<tr>
<td>Number of spray pumps in operation</td>
<td>All pumps in operation</td>
<td>1 pump in operation</td>
<td></td>
</tr>
<tr>
<td>Spray water temperature</td>
<td>Nominal temperature in the tank, nominal efficiency of the spray cooler</td>
<td>Maximum allowable temperature in the tank, minimum efficiency of the spray cooler</td>
<td></td>
</tr>
<tr>
<td>Spray system performance and its thermal efficiency</td>
<td>Nominal spray water flow, full heating-up of spray droplets</td>
<td>Minimum spray water flow, conservative estimation of spray thermal efficiency</td>
<td></td>
</tr>
</tbody>
</table>

199
Following comments should be made to the comparison of realistic and conservative assumptions as given in the Table 1:

- conservative analysis is always related to a specific criterion; many calculations, generally using different assumptions, are needed to check fulfilment of all acceptance criteria
- there is only one best estimate analysis, but many estimates of uncertainties are needed, i.e. a separate estimation for each criterion
- in some cases the conservative direction of a parameter is not obvious, e.g. underestimation of the heat removal at the beginning of a transient can increase the pressure initially, but it can also lead to early reactor shutdown with potential reduction of the containment peak pressure
- over-conservatism can not only influence significantly the value of a parameter to be checked against a selected criterion, but it can also produce misleading results from the point of view of timing for important phenomena

6. CONCLUSIONS

Considerable progress has been made in the field of accident analysis over the last two decades. Conservative computer codes often producing unrealistic results have been abandoned in favour of best estimate codes. Although conservatism will also in future remain as one of the safety principles in reactor design, there is a trend toward the broader use of best estimate accident analysis combined with evaluation of uncertainties. In line with increasing knowledge about modelled physical phenomena, capabilities of computer codes and computer hardware, there is also a continuous elimination of differences in methodologies used for design basis accidents and beyond design basis accidents as well as for analysis of particular postulated initiating events.

The IAEA is following this progress both in the development of revised Nuclear Safety Standards as well as in all other forms of support for its Member States. Co-operation between OECD and IAEA can render this process more effective.

7. REFERENCES


200


SESSION III - PART 1:
BEST-ESTIMATE METHODOLOGIES AND ASSOCIATED UNCERTAINTIES
OECD/CSNI Seminar on Best Estimate Methods in Thermal Hydraulic Safety Analysis
(Ankara, Turkey 29 June - 1 July 1998)

Best Estimate Methods in PWR Safety Analysis

by

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J P Rippon (NNC Ltd, UK) and M G Woodhill (Nuclear Electric Ltd, UK)

1.0 Introduction

This paper begins with an outline of the role in the UK licensing safety case for Sizewell B of a Probabilistic Safety Assessment (PSA), contrasting the Sizewell B approach with the construction of safety cases for other PWRs worldwide. The use of a PSA has stimulated the need for better codes and methods, particularly in the thermal hydraulics area, to model the outcome of very low frequency (severe) faults.

Two examples of the benefits obtained from application of better estimate methods in thermal hydraulic safety assessments for infrequent fault sequences are presented. The first relates to an assessment of the RCCA ejection fault for comparison with the Primary Circuit Integrity Limit. The second considers the implications of an uncontrolled RCCA bank withdrawal fault with respect to fuel clad failure considerations.

The paper goes on to describe for frequent fault assessments of Departure from Nucleate Boiling (DNB) how the use of 'better estimate' methods and assumptions can be used to underwrite the margins to safety limits based on the more traditional bounding approach.

2.0 Outline of Safety Case for Sizewell B

The Sizewell B licensing safety case includes a full probabilistic safety assessment comprising Level I (Design Basis), II (Beyond Design Basis) and III (Consequences). Whilst probabilistic safety assessments are now utilised in safety cases world wide, they are generally provided as an addition to a deterministic safety case. The latter considers initiating faults along with the most onerous single failure, plant unavailability due to maintenance etc. and utilises demonstrably conservative data and modelling. Where the safety case is supported with a probabilistic assessment best estimate approaches are followed. For Sizewell B, the fault sequences encompassed within the Design Basis were derived directly from the PSA. Several thousands of fault sequences output from the Event Trees were reduced by an extensive process of bounding, both between sequences and between initiating events, resulting in less than a 100 composite sequences for analysis. These are termed Bounding Limiting Design Basis Faults (BLDBFs). They comprise an initiating event (often one covered in an FSAR) and a number of additional plant failures, resulting in a fault sequence which typically has a frequency of occurrence of \(\sim 10^{-10}/\text{year}\)
or lower. Since any one BLDBF bounds a number of initiating faults, additional onerous plant conditions are often part of the specification. Whilst conservative representation of the core parameters (e.g. reactivity feedbacks, control rod worths, shutdown margins) are utilised, the modelling of plant and phenomena is generally ‘better estimate’.

An extensive program of code development verification and validation was undertaken prior to the start of the safety case analysis, as discussed in Section 3 below, and this enabled the complex and onerous BLDBFs to be analysed successfully.

Whilst the BLDBFs constitute the major part of the transient analysis, additional analyses are performed to address frequent faults, faults at shutdown, benign sequences and severe accidents.

**3.0 Thermal Hydraulics Codes used in the Sizewell B Safety Case**

For the Sizewell B PWR, the decision was made in the mid 1980’s to use advanced best estimate computer codes to perform the transient analysis supporting the Pre-Operational Safety Report (POSР). This was judged necessary to accommodate the complex and severe fault sequences derived from the PSA and also to satisfy the UK licensing authority’s desire relating to the use of more mechanistic modelling. This desire was strongly expressed at the Sizewell B Public Enquiry.

After an extensive review of the codes available world-wide in the mid 1980’s, it was decided that a number of Westinghouse codes then being developed most closely met the perceived requirements with the least risk that they could not be adequately developed, verified and validated in the stringent timescale necessary for the Sizewell B POSR.

The review had indicated a need for some further developments to these codes. These were carried out and then an extensive programme of verification and validation undertaken. These processes were carried out using rigorous quality procedures and at the conclusion of the programme a suite of verified and validated codes together with well defined user guidelines and a team of experienced code users was available to perform the Sizewell B POSR calculations.

**4.0 Examples of Better Estimate Methods in Infrequent Fault Analyses**

Two examples of the benefits obtained in infrequent fault analyses from application of better estimate methods are described below. The first relates to an assessment of the RCCA ejection fault for comparison with the Primary Circuit Integrity Limit. The second considers the implications of an uncontrolled RCCA bank withdrawal fault with respect to fuel clad failure considerations.
4.1 RCCA Ejection

The mechanical failure of a Control Rod Drive Mechanism pressure housing resulting in the ejection of a Rod Cluster Control Assembly (RCCA) and drive shaft out of the core is a severe fault extensively analysed in PWR safety cases. The accident results in a rapid rise in nuclear power which is arrested by the negative feedback resulting from the fuel temperature increase and the moderator density decrease. The fault provides a significant challenge to Fuel Structural Integrity and Primary Circuit Integrity Plant Limits. The latter will be considered here.

The sequence analysed for the safety case corresponds to the ejection of a control rod with the plant at hot zero power (HZP), and a number of additional failures including: failure of the first line of reactor trip; failure of an additional RCCA to insert after the reactor trip signal generated by the second line of protection; failure to open two pressuriser relief valves; and failure of all safety relief valves on the steam generators (SG’s) to open, the only relief available being due to the single Power Operated Relief Valve (PORV) on each SG.

Although a detailed 3D representation of the core was utilised in the analysis (193 assemblies represented by 1 radial and 20 axial nodes per assembly) conservatism was introduced into the analysis by increasing the ejected rod worth, reducing the delayed neutron fraction, increasing core bypass, reducing moderator temperature coefficient, and reducing fuel temperature feedback.

During the fault the core average nuclear power reaches approximately 5.2 times nominal full power before the fuel temperature reactivity feedback arrests the excursion. The resultant pressure transient seen by the RCS is compared with the Sizewell B Design Transient (DT) in Figure 1. The peak pressure variation is 0.53 MPa compared with the Design Transient value of 2.34 MPa.

The reason for the greatly reduced pressure transient (compared with the Design Transient) is due to the improved modelling capability in LOFT-5. The 3D neutronics capability and the ability to model 193 thermal hydraulic core channels provides a much more realistic representation of the RCCA ejection fault than that employed in the DT analysis. The DT calculations are based upon a 1D neutronics calculation which is used to provide a time dependent core average power to drive a thermal hydraulics model in which the core is split into 4 radial regions. The LOFT-5 3D neutronics calculation results in a reduced nuclear transient due to the more accurate representation of 3D feedback effects and the 193 thermal hydraulic core channel model provides a realistic estimate of the effects of the localised energy deposition due to the RCCA ejection. In the DT work, conservative assumptions had to be made to ensure the effect of localised energy deposition upon the thermal hydraulics had been adequately accounted for. Even in the 3D LOFT-5 analysis a large degree of conservatism has been retained by the use of conservative nuclear data. However, the benefits of more detailed thermal hydraulics representation is clearly demonstrated.
4.2 RCCA Bank Withdrawal

The fault to be analysed comprises the simultaneous withdrawal of 2 banks of RCCAs, without correct overlap, and at any rate up to the maximum mechanical rate possible. Typical FSAR analysis uses a point kinetics representation of the core when analysing the fault from power. A range of reactivity insertion rates are assessed and the effectiveness of the protection system in terminating the fault before all the DNB margin is eroded demonstrated. No single trip function provides adequate protection over the full range of reactivity insertion rates.

In the Sizewell B safety case, it is a requirement that at least two lines of protection preclude DNB for any reactivity insertion rate. Although this is readily demonstrated for the majority of insertion rates using the standard point kinetics modelling, difficulties have been experienced over a very limited range when due account of calorimetric uncertainties, RCCA shadow on the Ex-core detectors and perturbed reactor conditions due to automatic frequency response operation (AFRO) are taken into account.

Additional DNB margin has been obtained by using 3-D neutronics statepoint analysis at the limiting conditions to provide more realistic (but still conservative) axial and radial power profiles for use in the DNB analysis. Despite potentially onerous axial profiles possible in the RCCA bank withdrawal fault, the combined benefit of the realistic axial and radial profile has provided significant DNB margin even after due allowance for calculational uncertainties and the factors identified above. However, in this approach the neutronics and thermal hydraulics feedbacks are loosely coupled. To provide a more realistic assessment of the RCCA bank withdrawal fault LOFT-5 analysis utilising 3D neutronics and a 193 channel thermal hydraulics representation of the core has been carried out. Conservatism has been introduced into the analysis by increasing the RCCA bank worths by approximately 10%. (This impacts both the total reactivity release and the RCCA shadow effect on the Ex-core detectors, the response of the latter being represented in detail in LOFT-5). Results of this analysis have demonstrated the importance of the RCS thermal hydraulics effect on the transient. The realistic representation of the RCCA positions and reactivity release plus the realistic representation of the loop temperature response leading to a high Tin reactor trip results in the power transient being terminated by a reactor trip at a peak power value considerably lower than in the previous approach. A substantial margin to DNB is demonstrated.

A range of rod worths and withdrawal rates can be analysed to provide a similar study of protection response against reactivity insertion rate. However, the more realistic representation of the fault results in different degrees of effectiveness for the various lines of protection.

5.0 Departure from Nucleate Boiling Assessments in Frequent Faults

For Sizewell B, the design basis for normal operation, operational transients and frequent fault transients (faults with an initiation frequency \( \geq 1 \) in \( 10^3 \) years) requires protection against Departure from Nucleate Boiling (DNB). This is achieved by defining a limiting
value for the DNB Ratio (DNBR), which includes appropriate allowances for uncertainties. This limiting value is defined as the DNBR Safety Analysis Bounding Limit (SABL).

The basis of this DNBR SABL is the Revised Thermal Design Procedure (RTDP), first formulated by Westinghouse. The SABL takes account of statistical uncertainties in plant operating parameters, nuclear and thermal parameters, fuel fabrication parameters, computer codes and the DNB correlation. The value is selected on the basis of ensuring that there is a 95% probability that DNB will not occur, with 95% confidence, for the most limiting fuel rod in the most limiting fault that may occur in the design basis.

Typically, for Sizewell B, the coolant inlet temperature uncertainty due to instrument calibration errors is conservatively set at 1.1°C; appropriate allowances are made for other plant uncertainties. The variabilities that contribute to $F_{AH}^E$ (engineering factors) and $F_{AH}^N$ (nuclear factors) are also included as uncertainties. A total of seven state points are defined for the RTDP, four corresponding to core limits and three to faults, with adjustments to boundary conditions, where required, to ensure minimum DNBRs remain close to the SABL limit.

Additional margins are included as systematics. This results in a DNBR SABL of 1.345 for use with the Westinghouse WRB-1 correlation.

5.1 Lead Pin Assessment

The limiting DNBR is employed in deterministic calculations for the lead pin using the limiting (SABL) $F_{AH}$ value. The calculations are necessarily bounding, although credit is taken for some of the uncertainties already included in the derivation of the DNBR limit, to avoid double counting.

Results for a typical cycle in Sizewell B are shown in the Table below:

<table>
<thead>
<tr>
<th>Fault</th>
<th>Minimum DNBR</th>
</tr>
</thead>
<tbody>
<tr>
<td>Total loss of forced coolant flow</td>
<td>1.428</td>
</tr>
<tr>
<td>Locked rotor</td>
<td>1.354</td>
</tr>
<tr>
<td>Closure of all MSIVs</td>
<td>1.674</td>
</tr>
<tr>
<td>Loss of main feed to all steam generators</td>
<td>1.784</td>
</tr>
</tbody>
</table>

5.2 Whole Core Census

Although the lead pin approach gives confidence that DNB will not occur in frequent faults, the analysis can be interpreted as only demonstrating that there is a 95% probability that the lead pin does not enter DNB, with potentially a large number of rods close, in DNB terms, to the lead pin. This was identified as a potential concern and the solution has been to undertake a whole core DNB census, but using less conservative boundary conditions. The assessment criterion (no fuel failures in DNB) is met conservatively by showing that the
expected number of rods in DNB for any core condition within the protection envelope is less than one.

In practice, the expected number of rods in DNB is calculated at a set of statepoints which are representative of the worst reactor conditions that may be encountered. The statepoints are chosen to represent the overall trip protection envelope for Sizewell B. An implicit part of this choice (and indeed for all DNB analyses for Sizewell B) is the assumption that brief periods of DNB, eg., in the period after a trip point is reached but before the heat flux has been reduced by the trip, do not result in failure of the clad.

The assessment begins with a core-wide DNBR census for a given core state using the as calculated $F_{\text{AH}}$ value for each pin in the core. These results are then combined with parameter and correlation uncertainties to obtain a probability of any pin being in DNB.

In more detail the method is as follows. The uncertainty parameters are divided into local and core-wide classes. The local parameters are $F_{\text{AH},N}$ and $F_{\text{AH},E}$, and the core-wide parameters include core flow rate, inlet temperature, core power, system pressure, sub-channel code uncertainty and DNB correlation.

For a given group of fuel rods in the core, each of which has the same DNBR value, the probability of any pin in that group being in DNB can be calculated by convoluting the local and core wide uncertainty distributions; the sensitivities at the state points are calculated with an appropriate sub-channel code.

The expected number of pins in DNB, at that given statepoint, is then a simple sum, over all groups of pins, of the product of pin failure probability and pin numbers. This analysis is repeated for all the identified statepoints and the limiting condition determined. Typically, for Sizewell B, the results of this whole core census predict an expectation of around 0.1 pins in DNB for the most limiting condition.

Comparison with the results of the deterministic calculation in Section 5.1 highlights the benefits of the statistical treatment of uncertainties in the whole core census and the use of realistic $F_{\text{AH}}$ values. Thus, it can be seen from the lead pin results that the locked rotor transient exhibits a minimum DNBR of less than 1% greater than the limiting value, which could be interpreted as predicting close to one rod in DNB. This result compares with the whole core census prediction of <0.1.

6.0 Summary

A review has been presented of the Sizewell B safety case, outlining the role of a PSA to define the fault sequences encompassed within the Design Basis. The use of a PSA has stimulated the need for better codes and methods. Examples are presented of the benefits in thermal, hydraulic safety assessments from the application of better estimate methods in both infrequent and frequent fault analyses.
Figure 1

A Comparison of POSR and Design Transient RCS Pressure Variations
Use of Best-estimate Methods in a licensing case of 1300 MWe PWR

A. Amri (Institute for Protection and Nuclear Safety, Safety Evaluation Department, France)

ABSTRACT

The French utility EDF and the French designer Framatome have a clear tendency to use Best-Estimate computer codes (CATHARE, FLICA,...) in the safety demonstration, for instance in the frame of new fuel management or for future advanced reactors with a specific methodology depending upon the nature of the accident, i.e. design basis accident or beyond design basis accident.

The EDF and Framatome methodology based on CATHARE computer code has been applied to limited cases of design basis accidents, namely to small and intermediate break LOCA (SIBLOCA) and the long term phase of LOCA in the frame of 1300 MWe French PWR cycle extention (Gemmes project). A similar methodology has been applied to SIBLOCA for the European Pressurized Water Reactor (EPR).

This paper is aimed to describe the principals of this methodology and its application in the frame of Gemmes project. Its assessment performed by the Institute for Protection and Nuclear Safety is detailed and some conclusions are drawn.

1. INTRODUCTION

The GEMMES project involves looking further into the accident analysis studies of the 1300 MWe series safety analysis report, in the hope of implementing extended cycles (around 18 months) and three-batch core management with 4% enrichment in uranium-235 and the presence of fuel assemblies containing gadolinium. This in-depth study performed in the frame of this project has revealed changes due to the following:

a - The study rules applicable to the design basis accidents:

A set of new rules for Design Basis Accidents (DBA) analysis has been introduced in the frame of the GEMMES project. In particular, these rules specify to take into account a delay for the operator intervention and to include the operator intervention phase as a specific part in the transient thermalhydraulic study. Concerning the delay for the operator action, the French safety authority requires that a sensitivity study must be carried out taking into account a delay spanning from 10 minutes to 20 minutes for actions performed in the control room, and from 25 minutes to 35 minutes for the actions performed on the spot.

When these rules are applied, this affects the LOCA studies in two ways:

• firstly, the long-term LOCA phase (in particular the switch to simultaneous injection in the hot and cold legs) is now part of the operator intervention phase of the transient (so called
phase C) and requires conservative hypotheses to be applied. In particular, this means using a margin of uncertainty regarding the decay heat and taking uncertainties into account regarding the boron concentration in the sumps. However, the long-term LOCA studies of the 1300 MWe series final safety analysis report, intended to specify the emergency procedures, did not allow a margin for decay heat or management uncertainties and contingencies regarding the boron concentration in the sumps.

- the intermediate break studies of the safety analysis report assume that external power supplies are lost as soon as the reactor scram occurs. According to this hypothesis, the safety criteria are met. Sensitivity studies carried out later with the first generation codes (FRARELAP or FRACAS and ACONDA), show that when the primary pumps shut down seven minutes after the safety injection signal has been given, the safety criterion regarding the maximum cladding temperature (1204°C) is no longer met. In the GEMMES project, this difficulty is removed by using the new calculation methodology outlined below.

b - The calculation methodology:

This methodology, jointly developed by the French utility EDF and the French designer Framatome, is based, in particular, on the use of codes termed as "realistic" such as CATHARE or TRAC, in an approach defined as being "realistic and deterministic". The method is "realistic" in the sense that it uses one or more "realistic" computer codes. It is "deterministic" because it:

- is intended to be conservative: using a realistic code such as CATHARE obviously involves uncertainty regarding the reactor physical models and structure which, if necessary, should be allowed for by introducing penalties into the code physical models without distorting the physical behaviour of the whole,

- attempts to consider the specific aspects of transients: the uncertainty regarding their results, which should be considered in relation to the safety or decoupling criterion to be met, mainly results from a few dominant sensitive parameters which can differ depending on the type of transient in question. The method is meant to take the type of accident into account,

- aims to be conservative overall: bearing in mind how difficult it is to quantify all the uncertainties, uncertainty analysis is limited to those which affect the dominant physical phenomena, ensuring an overall degree of conservativeness is retained.

In practice, the following must be carried out for each transient when using this methodology:

- identify the dominant parameters and models as well as the associated uncertainty,

- check that the specific degree of conservativeness associated with the deterministic calculation allows for the impact of uncertainties (if necessary, go so far as to introduce a penalty to cover the uncertainties of the computer code in question),

- carry out sensitivity calculations on a wider range of parameters or models in order to validate the two previous stages.

This approach has been developed in a general manner to study the design basis accidents and has been applied by EDF in the context of the GEMMES project to:

- the long-term phase of large break LOCAs, the short term aspect being dealt with by Framatome Evaluation Model 1 (ME-FRA1),

- the entire transient, including the long-term phase, as regards intermediate and small break LOCAs.
c - Management

In relation to the matter of three-batch core management with 3.1% enrichment in uranium-235, GEMMES management increases the value of the enthalpy rise factor $F \Delta H$ and the decay heat. As regards the latter, the increase becomes significant several hours after the reactor scram.

2. RECAP OF FILE PREPARATION AND ANALYSIS LIMITS

During the first meeting of 16 March 1994, EDF made a preliminary presentation of the new long-term LOCA methodology which it intends to apply initially to the 1300 MWe series during the GEMMES project studies.

While the file was being prepared, the utility issued memos on the methodology and how it was to be implemented to deal with long-term LOCAs, intermediate breaks and small breaks.

Regarding application of the new methodology to the study of intermediate breaks, it would be appropriate to emphasise that the utility followed a step-by-step approach, using all the available calculation results as the file progressed. Without going into detail, three stages can be distinguished:

a - Stage 1:

During the meeting on 24 March 1995, EDF presented an analysis demonstrating how the new methodology to deal with the intermediate and small break short-term phase conformed to Appendix K of 10 CFR 50 which principally concerned large breaks originally. This analysis is an interpretation of the applicability of the Appendix to intermediate and small breaks.

IPSN noted that while the decay heat curve in question is the ANS 71 + 20% curve, the analysis presented by the utility does not help to identify the discrepancies in relation to the interpretation of Appendix K for intermediate and small breaks.

However, this analysis does not cover the aspects associated with checking that the mesh and time increment of the code version used are suitable. Proof of suitability is an express requirement in §II.2 of Appendix K of 10 CFR 50. This aspect will be covered in more detail in this paper.

During the same meeting, EDF presented a study of an intermediate break with delayed shutdown of the primary pumps, performed using the CATHARE 2 code, and taking into account the ANS71 + 20% curve. This study concluded that the safety criteria were met, however, it did not consider the uncertainty associated with the interfacial friction coefficient ($\tau_i$) which governs the swelling level in the core at a given collapsed level; the concomitance of taking into account the uncertainty associated with interfacial friction and the ANS71 + 20% curve and with an enthalpy rise factor of 1.7, means that the maximum cladding temperature is exceeded for the worst-case scenario (4 inch break with delayed shutdown of the primary pumps 15 minutes after the safety injection signal is given). In this first stage, the utility substantiated its first study by maintaining that the "lack of a degree of conservativeness for core $\tau_i$ is amply compensated for by the excessive degree of conservativeness of ANS71 + 20%".
b - Stage 2

Subsequent to the sensitivity calculations made regarding core $\tau_c$, EDF proposed a new approach, based on taking into account the SERMA + 10% decay law (which gives lower values than ANS71 + 20%), with an enthalpy rise factor of 1.7 and a core $\tau_c$ halved to compensate for the lack of a degree of conservativeness on the standard value of the latter parameter. According to this hypothesis, the utility showed that the safety criteria are met for the case of a 4 inch break and primary pump shutdown 15 minutes after the safety injection signal is given. It should be emphasised that EDF did not carry out sensitivity calculations on the size of the break or on the time the primary pumps took to shut down, and re-worked the worst-case scenario, identified using the parameters adopted for the previous studies (4 inch break with primary pumps shutting down 15 minutes after safety injection signal given). It considered it to be pointless to repeat the sensitivity studies which identified the worst-case scenario using the new approach. EDF substantiates its stand by pointing out that "the maximum cladding temperature calculated using SERMA + 10% and by halving core interfacial friction coefficient (new approach) is well below that obtained using ANS71 + 20% and standard core interfacial friction coefficient (extremely conservative hypotheses)". Later on, EDF officially asked the safety Authority to approve adoption of the new methodology (CATHARE + SERMA + 10%) for long-term LOCAs and the intermediate break short-term phase, as an alternative to applying Appendix K of 10 CFR 50.

IPSN had many reservations about this new approach, essentially based on taking into account the SERMA + 10% law of decay heat and which provides a considerable improvement on the maximum cladding temperature. Indeed, this new approach, proposed right in the middle of the case study involves:

- waiving the regulations in force which require compliance with Appendix K of 10 CFR 50 (1974 edition) in favour of the ANS71 + 20% law as decay heat, for short-term LOCAs of any break size,
- using a decay heat law, the values and associated uncertainties of which have yet to be completely substantiated by the utility. It should be emphasised that these justifications were requested during the Standing Committee meeting on February 3, 1994. While the GEMMES file was being prepared, EDF indicated that the justification requested will be sent at the beginning of February 1996, in accordance with the deadline set by the Safety Authority.

c - Stage 3

Bearing in mind the reservations of IPSN as regards the use of SERMA + 10, the utility proposed, during a third stage, a fall-back solution which should, it claims, make it possible to comply with the safety criteria. This solution, presented during the review meeting of 20 December 1995, consists in lowering the enthalpy rise factor from 1.7 to around 1.65, while taking into account the ANS71 + 20% decay heat law and dividing core interfacial friction coefficient in half.

It should be noted that this fall-back solution is temporary, indeed, the utility intends to implement the CATHARE methodology associated with the SERMA law and its uncertainties, to LOCAs, regardless of the size of the reactor coolant system break, and regardless of the transient phase, after having proven the upper-bound nature of this decay heat law.

All these elements will be explained in detail in the following sections dedicated to analysing application of the methodology to dealing with long-term LOCA phases and the intermediate break short-term phase.
2. NEW METHODOLOGY FOR STUDYING DESIGN BASIS ACCIDENTS

2.1 Principle of the new methodology

Before deciding whether or not the new methodology should be implemented in large break long-term LOCA phases and intermediate and small break short-term phases, the general principles of this methodology should be reiterated and analysed.

The objective of a design basis accident analysis is to demonstrate, with a very high level of probability and reliability, that the potential consequences of this design basis accident remain acceptable.

In practice, this involves checking, for each type of transient, the specific safety or decoupling criteria which intrinsically contain a safety margin in relation to the radiological objectives associated with these transients.

Furthermore, simulating accident situations raises some difficulties, associated particularly with:

- the exact specification of the boundary conditions to be adopted to guarantee the conservative nature of the study. The hypotheses to be adopted for these boundary conditions can vary from one case to another depending on the physics of the transient and the criteria to be met,
- the simulation capabilities as regards rendering discrete, the numerical solution, the approximate representation and limited understanding of the physical phenomena,
- the practical constraints.

The last two points change in line with the state of the art in the relative fields and with the computer power.

To take into account these limitations associated with the simulation and in order to guarantee the upper-bound nature of the demonstration, two approaches can be envisaged:

- either, to calculate the most probable value of the limiting parameter (cladding temperature or DNBR, etc...) as accurately as possible in view of the means available ("realistic" computer code which reflects what is known about the physical aspects, the numerical resolution, the calculation capabilities etc.) and to assign an uncertainty value to it with a certain degree of confidence. The study results are then compared to the criteria, bearing in mind the associated uncertainties. This type of approach, usually based on a statistical approach, such as the CSAU method (Code Scaling Applicability and Uncertainty) developed in the United States, requires the parameters, the models and sensitive hypotheses to be determined accurately, and the uncertainties associated with them to be quantified. This approach requires a properly validated calculation tool and numerous studies often need to be carried out.

A realistic computer code is understood to be a computer code which aims to replicate the physical phenomena as accurately as possible.

Overall uncertainty, associated with the use of such a code, results from the following in particular:

- uncertainty of the computer code, related to the accuracy of the physical models,
- uncertainty associated with the experimental data on the basis of which the physical models are quantified and the code is validated in its entirety,
- uncertainty associated with the effects of scale between the validating experiments and the reactor,
- uncertainty relating to initial and boundary conditions,
- uncertainty resulting from the use of simplifying hypotheses or from the choices made by the user regarding calculation options.

In this approach, bearing in mind that it usually proves impossible to calculate the uncertainty factor in an exhaustive and rigorous fashion for most thermal hydraulic applications (due to the complexity of the phenomena involved and the complexity and number of correlations implemented), it is necessary to:

- choose the parameters to be adopted for the statistical combination of uncertainties,
- identify the key parameters and place them in order of importance according to the dominant phenomena in the transient being studied,
- use weighting factors for the parameters (effect of scale for instance) which are not adopted statistically, and the basic uncertainty of which has not been quantified.

It should be noted that the first two points raise several difficulties and are largely based on the opinion of experts. As for the final point, it is one of the major problems with applying the method.

- either adopt a deterministic approach into which the appropriate degrees of conservativeness are introduced at various stages in the calculation. Such an approach can, in turn, be adopted either by:
  - using a deterministic method (in the sense of Appendix K of 10 CFR 50, 1974 edition) in which the uncertainties are not quantified at all, and instead, degrees of conservativeness are introduced for all the items identified. These degrees of conservativeness are sometimes high, in particular in the LOCA assessment physical models. Up until now this approach has been the one most commonly used to prove the safety of French PWR units, for LOCA,
  - or by considering a deterministic and realistic method in which the initial and boundary conditions remain conservative, as in the previous method, but the physical models used are realistic. However, with this approach, it is appropriate that the basic physical models which influence the overall results be upper-bound (in terms of conservativeness), with the objective of the experimental results being to get as close to them as possible. The objective of this approach is to reduce the excessive degree of conservativeness and to benefit from better handling of the margins.

It is the latter method, jointly developed by EDF and Framatome, and which uses the CATHARE computer code, that the utility adopted to study certain design basis operating conditions, and that it applied to deal with the long-term LOCA phase and the intermediate and small break short-term phase.

EDF chose this method for several reasons:
- progress made in the thermal hydraulic field, in particular as regards two-phase flow,
- the availability of a pertinent experimental basis,
the existence of advanced and realistic calculation tools which have been appropriately validated in the physical domain.

In principle, the method proposed by the utility is based on the following four stages:

- **Stage 1: use of a realistic computer code**

  The utility chose to use the realistic CATHARE computer code in the context of applications covered by the GEMMES project because of its simulation pertinence and the ability to transpose it to the reactor case.

- **Stage 2: Identification and assessment of the uncertainties on the dominant physical models**

  This stage consists in identifying the models which make the greatest contributions to the results and to the assessment of the uncertainties associated with them. Indeed, the utility considers that the majority of the uncertainty results from a few determining physical models which can differ depending on the dominant physical phenomena involved in each family of transients.

- **Stage 3: Checking the impact of uncertainties**

  This stage consists in checking that the impact of uncertainties on the determining physical models is lower than that which results from the expressly conservative margin which affected the deterministic calculation.

  Should the determining physical models of the computer code have been readjusted to an upper-bound value in relation to the experimental results adopted to qualify it, the utility considers that it is not necessary to quantify the impact of the uncertainty associated with these models, bearing in mind the readjustment in the systematically conservative sense of the model.

  Furthermore, although considerable discrepancies are observed between the results from studies allowing for uncertainties on the determining physical phenomena and those from deterministic studies carried out with expressly conservative margins, the utility considers that it is not necessary to quantify the contribution of the secondary parameters. If necessary, this contribution can be assessed by an upper-bound calculation where all these parameters are taken at their worst-case values.

- **Stage 4: impact of structure sensitivities and study hypotheses**

  This fourth stage is intended to ensure that the sensitivities on the results of various factors, in particular associated with the topological representation and the study hypotheses of the transient either do not call the conclusion from Stage 3 into question, or, depending on their nature, help to seek and fix an upper-bound configuration.

In practice, implementing the methodology for a given transient involves:

- checking that the computer code is appropriate overall,
- analysing and identifying the determining physical phenomena during the transient with regard to the parameter adopted as the safety or decoupling criterion,
- checking that the code is able to simulate these determining physical phenomena, in particular as regards the physical domain of validity and the qualification domain,
- identifying the key parameters of the code and assessing the associated uncertainty,
- carrying out sensitivity studies on the initial and boundary conditions,
- introducing penalties if necessary,
- making an overall conservative assessment.

2.2 Opinion of IPSN on the general principle behind the methodology

The new methodology is based on the use of a realistic code such as CATHARE, which aims to replicate the physical phenomena as faithfully as possible. The dominant physical models are conservatively readjusted in relation to the experimental results adopted for qualification. In principle, a calculation made using such a computer code will provide results which are on the conservative side, on the condition that it is valid for the transient in question. However, this conservativeness will be reduced in relation to the results obtained with the method used up until now.

The validity of a calculation code is checked on the basis of qualification-verification procedures, comparing the calculation results to a certain number of results from tests with separated effects or carried out on system loops.

For this purpose, the CATHARE computer code has a large qualification-verification program, operated on a basis of strict quality assurance and permanent monitoring by IPSN. Without going into detail, it should be noted that the CATHARE computer code uses a model with two fluids. The mass, impulse and energy conservation equations are averaged over space and time. It is also necessary to make a few simplifications on the basis of physical hypotheses and to establish the closure relations. These closure relations must be developed in order to translate the mass, impulse and energy transfers between each phase and the walls, and to the interface between the two phases.

The closure relations are developed and validated using a general methodology involving five main stages:

- analytical tests, including separated effect tests (interfacial friction, heat transfer at the walls and between phases etc.) and tests involving physical phenomena specific to certain reactor components, such as phase separation at the junctions for instance,

- development of a complete revision of the closing ratios (also called revision of the physical grids) on the basis of these tests. A revision of the closure relations is associated with one version of the code. A given revision can be associated with successive versions of the code. Indeed, a new revision of the physical grids contains new physical models, whereas a new version of the code uses new numerical methods, new modules or a new code architecture,

- analytical test qualification calculations in order to validate each of the closing ratios,

- calculations to check tests made on system loops in order to validate the general consistency of the closing ratios of the revision in question,

- issuing of the version of the code associated with a completely validated (qualified and checked) and fixed revision, accompanied by descriptive documents and the associated validation reports.
The long-term LOCA phase studies and the intermediate and small break short-term phase studies were carried out using Version 1.3, Revision 5 of the physical grids of the CATHARE code. The qualification/verification process for this version was carried out point by point on those models which had been modified in relation to the original version, CATHARE 1 from 1984.

An approach such as this is clearly not satisfactory in itself. Indeed, in the context of the design basis studies it would be preferable to use a fixed version of the code which has undergone the complete qualification/verification process.

Regarding the two uses proposed in the context of the GEMMES project, the code assessment made for these situations shows that the qualification of the code covers all the phenomena involved during the transient reasonably well. Moreover, the same assessment does not show excessively high or low results when checking the code on the basis of various tests carried out on system loops of different scales such as BETHSY, LOBI, LOFT, LSTF, PACTEL, PKL, PMK and SPES.

Bearing in mind the wide experimental basis and the sensitivity studies presented by the utility as regards the long term LOCA phase and the small and intermediate break short-term phase, IPSN has no particular reservations regarding the use of the CATHARE calculation code for these two specific cases. IPSN emphasises in advance that regarding the rules to be followed when using a realistic code in the safety studies, when they are applied across the board in the context of GEMMES to design basis accidents studies, they need their own version of the code and revision of the fixed physical grids, accompanied by a complete qualification/verification dossier.

For a study carried out with such a code, not taking into account the various uncertainties mentioned above, a certain number of precautions should be taken into account in order to obtain an upper-bound result. For this purpose, IPSN considers that as the code has been validated, implementation of the methodology becomes the predominant factor in the approach proposed by EDF.

Regarding this matter, IPSN notes that one of the stages of implementing the methodology consists in selecting, for a given transient and a criterion to be met, the determining physical phenomena, their dependence or their independence and any compensatory effects.

This choice, which governs the following stages of the method, is mainly made on the basis of expert opinion.

It should be noted that this limitation is not specific to the methodology proposed by the utility. Indeed, expert opinion has a role to play in practically all methods for assessing uncertainties associated with using realistic thermal hydraulic computer codes.

As this issue is a relatively important one in the proposed methodology, rational justification should be given and documented in order to establish the judgements of the experts underlying the choices of determining physical phenomena for a given type of transient and criterion to be met.

IPSN also notes that the utility does not expressly consider the uncertainties associated with the experimental data to which the results of the computer code are compared.

Indeed, uncertainties about the experimental results can be due to poor calibration and derivations in time or as a function of temperature. In addition, reading errors can be caused by the
measuring method itself. In the specific case of two-phase flow, it is not always possible to find the desired variable by measuring. For example, it is difficult to measure the water or steam temperature, or that of the structures or at the water-steam interface, as the two phases could appear at the same time. These uncertainties should not significantly call into question the principle of the methodology provided that they remain much smaller than the difference between the values calculated using the code and the average experimental values.

More generally, some open issues, not specific to this methodology, are pending. Indeed, it should be demonstrated that adding conservatism on dominant phenomena leads to an overall conservatism on the target parameter (maximum cladding temperature, for example). Moreover, it has to be checked that the dominant parameters approach covers the contribution of other neglected parameters. Another difficult issue is related to the scaling effects. In this respect, in the approach based on the dominant parameters, it should be demonstrated that it takes into account scaling effects, e.g. the conservatism demonstrated on the basis of small scale test facilities remains also for plant calculation (full scale).

With regard to this, it should be noted that work is underway both at the French Atomic Energy Commission (CEA) and the utility to specify an uncertainty assessment methodology in the context of using the CATHARE computer code industrially to study design basis accidents. This work should make it easier to assess the different items about which there is uncertainty and extend this method to other transients or to move it towards a more "realistic" approach. However, these elements are not determining in the context of an approach such as that proposed by the utility for dealing with the long-term LOCA phase and the intermediate and small break short-term phase. However, they will come into their own in the context of a more "realistic" approach.

In conclusion, IPSN considers that the methodology proposed by EDF is an interim approach between old conservative methodologies and best-estimate methodologies which include quantification of basic uncertainties and their statistical combination. This approach is acceptable in principle. Bearing in mind the wide experimental basis and the sensitivity studies presented by the utility, IPSN has no particular reservations regarding its implementation to deal with the long-term LOCA phase and the intermediate and small break short-term phase. The approach adopted for these cases remains conservative. However, IPSN emphasises that regarding the rules to be followed when using a realistic code in the safety studies, when they are applied across the board in the context of GEMMES to design basis accidents studies, they need their own version of the code and revision of the fixed physical grids, accompanied by a complete qualification/verification dossier.

Furthermore, IPSN noted that the utility does not specifically consider the uncertainties associated with the experimental data to which the computer code results are compared. However, these elements are not crucial in the context of an approach such as that proposed by the utility for dealing with the long-term LOCA phase and the short term intermediate and small break phase. However, they will have more importance in the context of a "realistic" approach. In the future, this method should benefit from the work currently being carried out at the French Atomic Energy Commission (CEA) and by the utility to specify an uncertainty assessment methodology in the context of using the CATHARE computer code industrially to study design basis accidents. This work should make it easier to assess the different uncertainty posts and extend this method to other transients or to move it towards a more "realistic" approach.
3. APPLICATION OF THE NEW METHODOLOGY TO THE LONG-TERM LOCA PHASE

This verification is made for the long-term LOCA phase and for the intermediate and small break short-term phase.

Application of the new methodology is described in detail for the long-term LOCA phase in order to clearly illustrate how the different stages are implemented. It was not considered necessary to go into the same amount of detail as regards dealing with intermediate breaks.

3.1 Nature of the problem

After a loss of coolant accident, the following long-term phenomena can be noted:

- in the case of a break in the cold leg, some safety injection water in the cold legs penetrates into the core, leaves it again as steam through the break, condenses in the containment and drains into the sumps. From a conservative standpoint, it can be assumed that the steam removed via the break does not contain boron, knowing that the safety injection water is borated, the boron concentration in the core increases (risk of jeopardising core cooling) and that of the sumps decreases (risk of re-injecting very diluted water into the core and causing it to go critical again). As a result, water must be injected into the hot legs before the boron crystallises in the core and before the boron concentration in the sumps drops below the critical core concentration. The water injected into the hot legs descends within the core, mixes with the highly borated water and reaches the break in the cold leg,

- in the case of a break in the hot leg, and during injection into the cold legs, the fluid which escapes directly through the break is a two-phase mixture; the boron concentrations in the core and the sumps therefore remain homogenous. When safety injection switches to the hot legs, some safety injection is maintained in the cold legs in order to ensure that decay heat is removed and to avoid initiating a core concentration/sump dilution process.

Simultaneous injection into the hot and cold legs makes it possible, once the switch to this injection mode has been made, to reduce the problem of core boron concentration and sump water dilution while ensuring that decay heat is removed, regardless of where the break is located.

In practice, this can be proved by checking the following points:

- that the boron has not crystallised in the core,

- that the core does not go critical again when switching to simultaneous injection into the cold and hot legs,

- that the capability of the safety injection system to remove decay heat without the risk of boron crystallising after switching to simultaneous injection is confirmed.

In conventional methodology, the last point was resolved by checking two decoupling criteria regarding the safety injection flow when switching over to simultaneous injection:

- a decoupling criterion regarding the flows injected into the cold legs for a hot leg break which should at least make it possible to remove the decay heat in the form of steam,

- a decoupling criterion regarding the flows injected into the hot legs for a cold leg break which must be such that the water injected reaches the core going upstream against the flow of steam produced to remove the decay heat. However, to simplify the proof, the possibility of the
injected water descending in counter current flow of the steam leaving the core is not taken into account. In practice, it is checked that the flow injected into the hot legs is able to condense all the steam produced and therefore to remove the decay heat in the form of a saturated liquid.

Checking all the criteria makes it possible to delimit a field of switching to simultaneous injection using a triangle called the switch-over triangle (given below) and to specify the time taken to switch to simultaneous injection as well as the boron concentration in the Refuelling Water System.

The base of the triangle gives the time beyond which the simultaneous safety injection flows can remove decay heat in the form of a two-phase mixture, regardless of where the break is located.

The left-hand side of the triangle specifies, as a function of the boron concentration in the Refuelling Water System, a limit time for switching, beyond which the boron concentration of the sump water could be too low to guarantee an effective multiplication factor of 0.99.

The right-hand side of the triangle gives, as a function of the boron concentration in the Refuelling Water System, a limit time for switching, beyond which the boron could crystallise in the core.

These lines are specified in view of the worst-case break locations and bearing in mind the margin on the boron concentration estimates.

Conventionally, and in particular in the P'4 series final safety analysis report, study of the long-term LOCA phase was used to help establish the operating rules applicable in the event of a LOCA and was not part of the accident study itself. The long-term LOCA phase was therefore effected while taking into account a certain number of realistic hypotheses, in particular decay heat (ANS71 with no margin) and the critical concentration taken without uncertainties. However, this former methodology presented a relatively high degree of conservativeness via the use of the decoupling criterion relating to the required safety injection flows into the hot legs imposing the removal of decay heat in water. This simplified hypothesis was due, at the time, to the lack of a two-phase computer code capable of processing the flow associated with this configuration.

The new rules for studying the design basis accidents that Electricité de France proposes to apply to the 1300 MWe units in the context of the GEMMES project would cause the long-term LOCA phase to be considered a part of Phase C of the accident and more conservative hypotheses would be applied to it, in particular as regards taking margins of uncertainty on decay heat, critical concentration and management contingencies into account.
It should also be noted that the new management system (extended cycles of around 18 months and three-batch core management with 4% enrichment in uranium-235) leads to a higher critical boron concentration.

Bearing in mind these changes, the switchover triangle cannot be specified when the former methodology is applied. Indeed, the switch to simultaneous injection must take place early on to meet the boron concentration limits in the sump and core water, the safety injection flows into the hot legs are therefore insufficient to guarantee removal of the decay heat in liquid.

In the context of the new methodology, Electricité de France proposes a more physical approach, based on the use of the CATHARE computer code to model the flow of water in liquid form in counter current to the steam, in order to replace the conventional decoupling criterion relating to flows injected in hot legs with a new criterion:

"the safety injection water injected in the hot legs descends into the core, reducing the concentration and removing the decay heat".

It should be noted that in the context of dealing with the long-term LOCA, the new methodology calls upon both the CATHARE computer code in its one-dimensional version and the realistic TRAC-PF1 code, in its three-dimensional version. Indeed, the hydraulic behaviour in the reactor vessel is largely associated with three-dimensional physical phenomena.

3.2 Implementing the methodology

The REFLET computer code is used to determine the authorised domain in the plane (Cb, switch over time), with regard to the boron crystallisation in the core criterion and the non-return of core criticality.

With the exception of the decay heat curve, which is discussed further on, the aggravating factors and the hypotheses adopted do not give rise to any specific comment.

Once this domain has been determined with regard to the boron crystallisation and non-return to core criticality criteria, it must be checked that the flows injected by simultaneous injection enable the decay heat to be removed and the core concentration to be reduced. This check is made by implementing the new methodology based on the use of the CATHARE code.

3.2.1 Determining physical phenomena

The criterion to be checked is the descent of the safety injection water injected into the hot legs in counter current to the steam, down into the core, so the governing physical phenomenon is that of the entrainment of water by steam in the hot legs and the steam generator water boxes. Indeed, this phenomenon limits the quantity of safety injection water which can enter the core, reduce its concentration and remove the decay heat.

The entrainment of water is directly affected by the velocity of the steam in the hot leg; this velocity in turn depends on other parameters such as power, pressure, the possible presence of water slugs in the intermediate legs which affects the loss of overall flow and the steam path cross-section (hence its velocity).
Other phenomena can play an important role on system effects:

- condensation of the steam by sub-saturated safety injection water. This condensation has two opposing effects: the sub-saturated water makes it possible to condense a part of the steam, thus reducing its flow, but it can occupy a part of the cross-section of the cold leg or intermediate leg passage, thus increasing the flow per unit volume of the steam, and hence its velocity,

- the mixture of safety injection water in the hot legs with the core water. In the core, the water and steam flowing against each other will generate multi-dimensional phenomena. Indeed, radially, there would be a zone where the safety injection inlet was located, which would be principally descending, with low void fractions, and a zone where the steam outlet was located, which would be principally ascending with high void fractions and there would be transverse flows between these zones. Axially, the void fraction would vary.

IPSN has no specific comments to make on the selection of the determining physical phenomenon, or on the other secondary phenomena which are limiting as regards the safety injection water injected into the hot legs penetrating into the core.

3.2.2 Verification of CATHARE code applicability

An EDF memo presents calculations from experimental tests conducted on the ECTHOR (medium scale) and UPTF (Scale 1) loops using the CATHARE code. These calculations show that for the physical conditions encountered in these situations, the CATHARE code predicts, in the hot leg, a lower counter-current limit (therefore authorises a lower counter-current liquid flow) than that determined with ECTHOR and UPTF. In this, the code is the upper-bound of the experimental results.

The multi-dimensional phenomena of the safety injection water injected into the hot legs descending into the core were confirmed by the UPTF, CCTF and SCTF test results, and were backed up by a reactor calculation using the TRAC code in which the vessel is modelled in three dimensions. The applicability of the code for these studies is based on the fact that it correctly assesses the three-dimensional phenomena observed in experiments in the core in recovery and post-recovery situations, these phenomena having the same origins (injection at the top of the core) in these two situations as they do in long-term LOCA situations.

Conventional modelling of the reactor vessel with the CATHARE code uses an axial element for the core and a volume element for the upper plenum. To try and simplify the problem, a single-volume model of the core and the upper plenum was adopted in compliance with recommendations made by the team responsible for developing the code when the three-dimensional hydraulic phenomena have to be represented in complex geometries.

It has been checked that this choice of model does not affect the system effects, by comparing, at the vessel inlet, the results provided by CATHARE and TRAC respectively, in the long-term LOCA phase.

The CATHARE code has been qualified with regard to the condensation of steam by sub-saturated safety injection water, on the basis of COSI tests. With regard to the existence and possible clearing of water slugs in the intermediate legs, it was qualified using ECTHOR BU tests. Qualification of the code with regard to these two phenomena was confirmed by the CATHARE code assessment made by the CATHARE technical committee.
To supplement the demonstration, IPSN asked the utility to extend the experimental basis for checking the code, in particular at the BETHSY 6.10 test regarding the large break, long-term LOCA phase. For this purpose, IPSN noted that the first post-calculations from this test, made by the BETHSY team show that the CATHARE code underestimates how much water is carried along by at least 20%, regardless of the injection configuration (cold leg and combined injection).

For the utility, the results from the BETHSY 6.10 test are not really representative of the reactor in long-term LOCA phases because of the geometry of the reactor vessel and the hot leg of the loop at the steam generator inlet which is not representative of the reactor geometry. The geometry of the BETHSY loop tends to favour water entrainment. These phenomena are being analysed and a supporting experimental program on the MHYRESA loop is planned for 1996. All things taken into account, the ECTHOR and UPTF geometries are the most representative of the reactor geometry.

Furthermore, if it is considered that the amount of water entrained is underestimated by 20%, flow into the core would then be enough to remove the decay heat and reduce the boron concentration in the core. Indeed, four hours into the accident, around 18 kg/s of water at saturation point are enough to remove the decay heat as latent heat, and several kg/s is enough to reduce the boron concentration in the core, the injection flow into the hot leg being 180 kg/s.

In conclusion, bearing in mind the points made by the utility in the methodology and design memos, the importance of qualification and validation by experiment, and the supplementary material presented during technical meetings, confirmed by the CATHARE code assessment report, IPSN has no comments to make either on the new method proposed by the utility to determine the time needed to switch to simultaneous injection in the event of a LOCA, or on the suitability of the CATHARE computer code for simulating the determining physical phenomena involved during a long-term LOCA.

3.2.3 Introducing penalties

Bearing in mind the above sections and the fact that entrainment of water is the governing physical phenomenon for this transient and that it is dealt with conservatively by the CATHARE code (this remains to be checked on the BETHSY setup), it is not necessary to introduce a specific system of penalties.

IPSN agrees on this conclusion.

3.2.4 Choice of initial and boundary conditions

In addition to the elements mentioned above, the utility has carried out a certain number of sensitivity studies in order to justify the main hypotheses adopted and, if necessary, to quantify their impact.

It has drawn the following conclusions from these sensitivity studies:
- sub-saturation of safety injection water is a beneficial phenomenon overall, with regard to the criterion to be met. Indeed, this sub-saturation reduces the quantity of decay heat to be removed in the form of steam and hence the flow of steam emanating from the core. As a result, the study considers that the safety injection water is at saturation,
- the increase of safety injection flows is favourable on the transient because it leads to a reduction of the steam flow emanating from the core and speeds up the reduction of core
concentration. It is therefore penalising to consider a minimum safety injection flow. With regard to this, the single aggravating factor rule applies here, supposing that a single safety injection train is available. This aggravating factor is penalising for a four-loop reactor (1300 MWe or N4 series) because on these reactors, the safety injection configuration is such that there is only one hot leg connection left if only one safety injection train is available. The existence of cold water slugs in the intermediate legs of the other loops tends to maximise the steam flow and therefore leads to the safety injection water being carried along.

- it is not necessary to begin the transient with seals in the intermediate legs; indeed, these seals, the existence of which maximises the speed of the steam by blocking a part of the passage cross-section, are rapidly removed. This conclusion is supported by sensitivity calculations with different water inventories in the undamaged loops and the broken loop and confirmed by tests carried out on the BETHSY loop,

- the position of the break in cold leg is the worst-case position with regard to overconcentrations of boron in the core. This aspect is confirmed by a study carried out with a break located on the intermediate break at the steam generator outlet,

- the lowest possible primary pressure is chosen, that is to say 1 bar. Indeed, the higher the primary pressure, the greater the density and enthalpy of the steam and the lower the velocity at which the steam escapes from the core for a given decay heat,

- the higher the core pressure, the greater the flowrates of steam emanating from the core, and the greater its velocity. With regard to this, EDF made its calculations using the SERMA + 10% decay heat curve; the use of this curve will be discussed later on.

The sensitivity studies carried out by EDF and the associated interpretations do not call the choice of these initial and boundary conditions into question. Moreover, the concomitance of these hypotheses does not, in theory, lead to compensatory effects. As a result, IPSN considers that, apart from the decay heat curve, the values of which and the associated uncertainties remain to be confirmed, this choice of initial and boundary conditions is conservative with regard to entrainment of water in the hot legs.

3.2.5 Overall conservative assessment

It is clear in the preceding paragraphs that the overall assessment is, provided the general reserves expressed in the § 2.2, conservative because:

- the code correlations, in particular those relating to water entrainment in the hot legs, are conservative in principle,

- the compensating effect is limited by the limited number of the dominant parameters (mainly the water entrainment in the hot leg),

- the initial and boundary conditions are chosen, on the basis of sensitivity studies, with a conservative leaning,

- the rules and hypotheses of the studies are conservative (application of the single aggravating factor to a safety injection train, uncertainties taken into account on the decay heat as well as on the boron concentration in the sumps).

Bearing in mind all these points, IPSN considers that it would be acceptable to implement the new methodology in the long-term LOCA phase.
However, in implementing it, EDF uses the SERMA + 10% curve, the values of which and the associated uncertainties have not yet been completely substantiated. A change as regards the decay heat values does not call the basic methodology into question, however it has a significant impact on the time taken to switch to simultaneous injection. The study of the long-term LOCA phase in the context of the GEMMES project, carried out using the SERMA + 10% decay heat curve, requires a switch to simultaneous injection within nine hours after the accident begins, for a maximum boron concentration in the Refuelling Water System of 2500 ppm (in comparison with the delay of 14 hours and a maximum boron concentration of 2150 ppm for the current management system).

If the decay heat values were called into question, this would modify the maximum time allowed; the time would be shorter if the decay heat was higher. A calculation made by EDF with a decay heat of 40 MW, corresponding to a time of 4 hours after the beginning of the accident, shows that the overall behaviour of the system is the same as in the reference case (decay heat of 32 MW corresponding to a switch-over time of around nine hours) and that the criteria are met. Thus, the conclusions of the study should not be called into question by a change in the decay heat values which, in any case, will be similar to the current values.

It should be noted that in current practice, these studies were carried out using the values from the ANS71 decay heat curve without a margin. For the cooling time we are interested in here, these values are slightly lower than the SERMA + 10% decay heat curve values.

In conclusion, IPSN considers that the application of the new long-term LOCA phase assessment method is acceptable. The two points currently being looked at (decay heat and water entrainment underestimated in Test 6.10 on the BETHSY loop) are not of a nature to call into question the times for switchover to simultaneous injection of between four and nine hours specified by EDF.

4. APPLICATION OF THE NEW METHODOLOGY TO INTERMEDIATE AND SMALL BREAKS

4.1 - Position of the problem

In this category of reactor coolant system breaks, the limiting case is an intermediate break with delayed shutdown of the primary pumps. Indeed, maintaining the primary pumps in operation while guaranteeing proper core cooling in the short term, means a considerable flow is maintained at the break and the main coolant system drains faster. The shutdown of the primary pumps therefore considerably lowering the water in the reactor vessel and in the core with the possibility of partial uncovering of the latter. Furthermore, when it is steam flowing through the break, the reactor coolant system loses pressure faster, which makes it possible to start injection from the accumulators and to re-cover the core. Therefore there is a window of time where shutting down of the primary pumps is critical with regard to core over-heating. For a given break size, the worst-case conditions for core uncovering will come into play when the primary pumps have shut down sufficiently late to minimise the primary-side water inventory, but early enough to maximise the delay between when the primary pumps shut down and the accumulator injection pressure reaches its threshold.

Core uncovering subsequent to lowering of the water in the reactor vessel, following shutdown of the primary pumps appears to be the governing factor with regard to core over-heating. The consequences of this uncovering on core heating will depend on the level of swelling in the core.
after lowering and on how long uncovering lasts. The swelling level is determined by the quantity of water which manages to fall into the core and by the average void factor in the zone which remains covered. The descent of the water from the upper parts of the system back down to the bottom of the reactor vessel is governed by the interfacial friction models in the hot legs, the hot leg nozzles, the upper core plate and in the core. Interfacial friction involving the upper core plate and in the core is the most inhibiting.

The maximum cladding temperature (1204°C) criterion may be exceeded in the event of an intermediate break with delayed shut-down of the primary pumps if conventional methodology, based on the FRACAS and ACONDA computer codes is used to carry out the calculation, and if the new design basis accidents study rules are adopted for 1300 MWe units in the context of the GEMMES project. These include, in particular, an operator action time which may extend up to 20 minutes, and the time the enthalpy rise factor takes to increase from 1.55 to 1.70. It should be recalled that even with an enthalpy rise factor of 1.55, these first generation codes, the physical models of which contained certain limitations, request the primary pumps to be shut down no more than seven minutes after the safety injection signal has been given in order to comply with the safety criteria.

This is why EDF proposes that the "deterministic and realistic" methodology, based on the use of the CATHARE calculation, be applied to intermediate and small break transients.

4.2 Opinion of IPSN

The file is compiled in three stages which were covered in §2.1.

The following conclusions can be drawn from implementing this methodology and the many sensitivity studies associated with this. They are related to the interfacial friction model, which determines the swelling level using the coefficient of interfacial friction in the core, and the choice of position for the pressuriser on the damaged loop in a two-loop model, both tend to minimise the cladding temperatures during a transient.

As regards estimating the swelling level, the impact of which is a 60°C increase in the maximum cladding temperature, EDF has eliminated it by introducing a penalty to the interfacial friction coefficient by dividing it in half.

The position of the pressuriser on the damaged loop introduced a limited increase of around 20°C to the maximum cladding temperature. This model, chosen to save calculation time, does not call the general conclusions of the study into question because this increase can be taken into account when the study results are compared with the criterion.

During the transient, the water accumulates in the intermediate legs which block the path of steam towards the break. This water slug is removed by readjusting the pressure between the core (drop in level) and the rising part of the intermediate legs (removal of the slug via the break). As this phenomenon is purely gravitational, the choice of mesh (number and size of links), in particular in the intermediate legs, it is a parameter which has a very notable effect on the simulation results.

Indeed, on the basis of sensitivity calculations, for a 3 inch break in the cold leg on a 900 MWe reactor, the average fuel rod cladding temperature increase can be estimated at 55°C (in other words an increase of around 85% on the cladding temperature of the hot fuel rod) when switching from a large mesh (9 links for the cold legs with loops intact, 15 links for the cold leg of the
damaged loop and 7 links for the intermediate legs with one link only at each bend) to a finer mesh (15 links for the cold legs with loops intact, 29 links for the cold leg of the damaged loop and 14 links for the intermediate legs with 3 links at each bend).

- EDF adopted a fine mesh as the reference mesh for its intermediate break studies without introducing a cladding temperature penalty. This choice was justified by the fact that this fine mesh was theoretically developed for a correct description of the geometry of the intermediate legs, which is not possible with the simplified mesh.

Furthermore, at the request of IPSN, EDF gave further justification relating to the choice of reference mesh. These reasons are the following:

- the CATHARE code development team recommends using at least three links in the bends, which corresponds to the reference mesh adopted. This recommendation stems from the sensitivity calculations made on the ECTOR and IVO loops,

- the phenomena sensitive to the intermediate loop model with the loop slug clearing phenomenon, which is only significant when the reactor coolant system is being drained in stratified flow with low or virtually no loop flow. This situation arises for small break and intermediate break transients with early shut-down of the primary pumps, which are not limiting compared to the scenarios with delayed shut-down of the primary pumps, for which this situation does not arise, in the short term at least.

In view of this, EDF considers that there is no need to introduce a penalty to the cladding temperature due to the choice of reference mesh.

IPSN considers the first argument to be acceptable. Regarding the second point, studies carried out at IPSN using the CATHARE code for a 3 inch break in the cold leg of a 900 MWe reactor with primary-side pump shut down at 15 minutes, does not prove clearly enough the phenomenon of formation and clearing of the water slug in the critical time slot with regard to core uncovering (a few minutes after the primary pumps shut down). In the longer term, these same studies show that this phenomenon can occur, but at a much slower pace which does not result in core uncovering.

As regards mesh suitability, EDF considers that with the exception of the downcomer, which is not the seat of dominant phenomena for intermediate breaks, the zones of the reactor coolant system and the steam generator secondary system which use the "pipe" model of the CATHARE code, have been substantiated by sensitivity studies.

IPSN points out that, as it has been demonstrated above, the cladding temperature is sensitive to the loop mesh, in particular that of the intermediate branches. To apply §II.2 of Appendix K, it would have to be proven that the maximum cladding temperature remains stable if the reference mesh is finer. However, bearing in mind the answers given by EDF and the nature of the problem, the reference mesh adopted is acceptable for this intermediate break study with delayed shut-down of the primary pumps.

Nonetheless, the choice of mesh would need to be substantiated and its suitability proven should this CATHARE code-based method be extended to the study of other transients.

As regards the time increment, EDF indicated that the studies were carried out with a maximum time increment of 0.5 seconds, which is below the 1 to 2 seconds recommended by the CATHARE team for slow transients. In addition, a check is made of the mass and energy values,
which are included in the code read-outs and are analysed to assess the accuracy of the results obtained. If there are unexplained physical phenomena, a sensitivity study can be carried out on the maximum time increment given as input data. EDF has specified that such cases are not presented either while the generic sensitivity study file is being compiled, nor during applications specific to the GEMMES study.

Bearing these elements in mind, IPSN has no particular comments to make on the suitability of the time increment in the context of a study on intermediate and small breaks.

Concerning decay heat values, one can expect that they have a considerable influence on the maximum cladding temperature. Indeed, all things being equal, the switch from the ANS71 + 20% decay heat curve to the SERMA + 10% decay heat curve, results in an increase in the maximum cladding temperature which depends on the moment the core becomes uncovered and which can reach 200°C.

IPSN has no particular reservations as regards the final implementation of the methodology for intermediate and small break studies. Indeed, at the request of IPSN, and in compliance with Appendix K, EDF continued its study using the ANS71 + 20% decay heat law, a penalty on interfacial friction, and an enthalpy rise factor of 1.65 as well as other conservative factors on the initial and boundary conditions and the structure, resulting from various sensitivity studies, and concluded that the safety criteria are complied with for the limiting case: 4 inch break and delayed shut-down of the primary pumps, 14 minutes after the safety injection signal. It also used sensitivity coefficients to estimate that these conclusions remained valid for an enthalpy rise factor of 1.67.

In view of this, IPSN considers implementation of the new methodology for intermediate and small breaks in the GEMMES project context, to be acceptable. The enthalpy rise factor must remain below 1.67 to ensure that the safety criteria are met.

5. CONCLUSION

As regards the new "deterministic and realistic" methodology, based on the use of the CATHARE code, and implemented by EDF to deal with long-term LOCAs and the short-term intermediate and small break phase:

- IPSN considers that the methodology proposed by EDF is an interim approach between old conservative methodologies and best-estimate methodologies which include quantification of basic uncertainties and their statistical combination. This approach is acceptable in principle. Bearing in mind the wide experimental basis and the sensitivity studies presented by EDF, IPSN has no particular reservations regarding the implementation of this methodology to deal with long-term LOCAs and the short-term intermediate and small break phase. The approach adopted for these cases remains conservative. However, IPSN emphasises that regarding the rules to be followed when using a realistic code in the safety studies, the generalisation of this use, to design basis accidents studies in the context of GEMMES, requires a frozen version of the code and of revision of the physical grids, accompanied by a complete qualification/verification dossier.

More generally, some open issues, not specific to this methodology, are pending. Indeed, it should be demonstrated that adding conservatisms on dominant phenomena leads to an overall conservatism on the target parameter (maximum cladding temperature, for example). Moreover, it has to be checked that the dominant parameters approach covers the contribution
of other neglected parameters. Another difficult issue is related to the scaling effects. In this respect, in the approach based on the dominant parameters, it should be demonstrated that it takes into account scaling effects, e.g. the conservatism demonstrated on the basis of small scale test facilities remains also for plant calculation (full scale).

Furthermore, IPSN noted that the utility does not specifically consider the uncertainties associated with the experimental data to which the computer code results are compared. However, these elements are not crucial in the context of an approach such as that proposed by the utility for dealing with the long-term LOCA phase and the short term intermediate and small break phase. However, they will have more importance in the context of a more "realistic" approach. In the future, this method should benefit from the work currently being carried out at the French Atomic Energy Commission (CEA) and by the utility to specify an uncertainty assessment methodology in the context of using the CATHARE computer code industrially for design basis accidents analysis. This work should make it easier to assess the different uncertainty posts and extend this method to other transients or to move it towards a more "realistic" approach.

- IPSN considers that the application of the new long-term LOCA phase assessment method is acceptable. The two points currently being looked at (decay heat and water entrainment underestimated in Test 6.10 on the BETHSY loop) are not of a nature to call into question the time for switchover to simultaneous injection in both hot and cold legs, specified by EDF for 1300 MWe plants in the frame of GEMMES project.

- IPSN considers implementation of the new methodology for intermediate and small breaks in the GEMMES project context to be acceptable. The enthalpy rise factor must remain below 1.67 to ensure that the safety criteria are met.
Application of Best Estimate Methods to LOCA in a PWR

OECD/CSNI Seminar, June 29th - July 1st 1998, Ankara, Turkey

F. Depisch, G. Seeberger, S. Blank Siemens KWU
Table of Contents

1. Introduction
2. Scope of LOCA Analysis for PWR 1300 MW of KWU Design
3. Methodology for LB-LOCA Analysis
   CSAU Approach
   - Element 1: Requirements and Code Capabilities
     » Selection of Code System RODEX, S-RELAP5, COCO
     » PIRT
   - Element 2: Assessment and Ranging of Parameters
     » Assessment Matrix for S-RELAP5
     » Results of Validation of S-RELAP5
     » PWR plant model with S-RELAP5
   - Element 3: Sensitivity and Uncertainty Analysis
     » Sensitivity studies
     » Treatment of Plant, Code and Fuel Related Uncertainties
     » Total Uncertainty

OECD/CSNI Seminar 6/7 1998, Ankara, Turkey
1. Introduction

- PWR: 1300 MWe 4 Loop NPP designed by Siemens KWU
- ECCS consists of 4 subsystems with 2 Accus, 2 RHR and 1 SIP Pump each
- ECCS injection in hot and cold leg
- ECCS acceptance criteria:
  - PCT < 1200 °C
  - d (ox) < 17 %
  - V (Zirc,ox) < 1 %
2. Scope of LOCA Analysis for a 1300 MW KWU PWR

- Small and Medium Breaks:
  - 50 cm² up to 442 cm² in cold leg
  - 21 cm² up to 437 cm² in hot leg
  - 20 cm² in RPV bottom
  - PCT 609 °C for 442 cm²
  - Deterministic Analysis

- Large Breaks:
  - 2*A in cold leg, hot leg, between SG and MCP
  - In cold leg 2*A down to 2* 553 cm²
  - Statistical Analysis for Worst Case
3. Methodology for LB-LOCA Analysis

- Siemens approach for LB-LOCA follows CSAU Methodology developed by USNRC
- Methodology developed for Non KWU 3/4 Loop PWR’s in 1993 (SPC)
- Methodology adapted to KWU 4 Loop PWR by Siemens Germany in 1996/1997
- Characteristics of CSAU Methodology
  - Use of Best Estimate Code System
  - Quantification of Uncertainties
  - Stepwise Approach in 3 Elements
Code Scaling, Applicability and Uncertainty (CSAU) Evaluation Methodology
CSAU Element #1: Requirements and Code Capabilities

- Specification of Scenario, Selection of NPP
  - LB LOCA
  - 1300 MW KWU PWR
- Selection of Code System
  - RODEX, S-RELAP5, COCO
- Phenomena Identification and Ranking Table (PIRT)
CSAU Element #1: Requirements and Code Capabilities

- Selection of Code System: RODEX, S-RELAP5, COCO
- S-RELAP5:
  - Simulation of thermohydraulic behavior of NPP during transients
  - Based on RELAP5/MOD2 and MOD3.0
  - S-RELAP5 Model Improvements by Siemens:
    » Multi Dimensional Capability for DC, UP, LP and core
    » Treatment of Non Condensables
    » Interface Friction and Mass Transfer
    » Heat transfer Correlations
    » CCFL
    » 2 Phase pump degradation model
    » RODEX3 fuel model
    » Choked flow
CSAU Element #2: Assessment and Ranking of Parameters

- Assessment Matrix for development and application of S-RELAP5
  - 2 levels of assessment:
    » development of code S-RELAP5
    » assessment of finalized code S-RELAP5
  - Test facilities for development: ORNL THTF, EPRI pump tests, GE level swell, Marviken, Bennett, Zorita, Riso, Halden
  - Test facilities for assessment: LOFT, UPTF, FLECHT-SEASET, CCTF, PKL-II
### PWR LBLOCA Process Identification and Ranking with Experiment (1)

<table>
<thead>
<tr>
<th>TEST FACILITY &amp; TEST NUMBER</th>
<th>PHENOMENON</th>
<th>COMPONENT</th>
</tr>
</thead>
<tbody>
<tr>
<td></td>
<td>Stopped energy</td>
<td>Oxidation</td>
</tr>
<tr>
<td>CCF-79</td>
<td>X</td>
<td>X</td>
</tr>
<tr>
<td>PALL B2</td>
<td>X</td>
<td>X</td>
</tr>
<tr>
<td>Halden</td>
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<td>X</td>
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<td>Risø</td>
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<td>ZöRFA</td>
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<tr>
<td>BENNETT</td>
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<td>X</td>
</tr>
<tr>
<td>MARKKEN</td>
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<tr>
<td>MARKKEN</td>
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<tr>
<td>GE Will</td>
<td>X</td>
<td></td>
</tr>
<tr>
<td>CEARA 20 Pumps Tests</td>
<td></td>
<td></td>
</tr>
<tr>
<td>OOPAL THF</td>
<td>X</td>
<td>X</td>
</tr>
<tr>
<td>UPP</td>
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</tr>
<tr>
<td>Flich-Seasat</td>
<td>X</td>
<td>X</td>
</tr>
<tr>
<td>Flich-Seasat</td>
<td>X</td>
<td>X</td>
</tr>
<tr>
<td>CCF</td>
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<td>X</td>
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<tr>
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<td>X</td>
</tr>
<tr>
<td>LOFT</td>
<td>X</td>
<td>X</td>
</tr>
</tbody>
</table>
CSAU Element #2: Assessment and Ranging of Parameters

- PWR Plant model with S-RELAP5
  - 4 loops modelled
  - 2D component in DC, Core and UP
  - core model for ECCS injection into hot legs
  - modelling of complete ECCS system
CSAU Element #3 : Sensitivity and Uncertainty Analysis

- Sensitivity studies with code system RODEX, S-RELAP5, COCO
  
  - Criterion for sensitivity : PCT
    » PCT most sensitive output parameter, rest of acceptance criteria prop. to PCT
  
  - 2 types of sensitivity study :
    » definition of bounding values of deterministically treated input parameters
    » quantification of sensitivity of statistically treated input parameter
CSAU Element #3: Sensitivity and Uncertainty Analysis

- Sensitivity studies with code system RODEX, S-RELAP5, COCO
  - 13 Studies to define bounding values of deterministically treated parameters:

<table>
<thead>
<tr>
<th>Parameter</th>
<th>Bounding Value</th>
</tr>
</thead>
<tbody>
<tr>
<td>break location</td>
<td>RCP-RPV</td>
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<tr>
<td>break size</td>
<td>2A</td>
</tr>
<tr>
<td>axial power profile</td>
<td>top skewed</td>
</tr>
<tr>
<td>radial power peaking</td>
<td>1.6</td>
</tr>
<tr>
<td>reactor kinetics</td>
<td>low feedback rho-void, strong for Doppler neglected</td>
</tr>
<tr>
<td>swelling rupture of fuel rod</td>
<td>2 hot accus, 1 SIP</td>
</tr>
<tr>
<td>single failure, repair case</td>
<td>at turbine trip</td>
</tr>
<tr>
<td>loss of offsite power</td>
<td>0.001 sec</td>
</tr>
<tr>
<td>time step in S-RELAP5</td>
<td></td>
</tr>
</tbody>
</table>
• Statistically Treatment of Uncertainties:
  - Separate treatment in 3 different groups
  - Uncertainties of plant parameters (low sensitivity)
  - Uncertainties related to fuel (high sensitivity)
  - Combination via Monte Carlo Method
CSAU Element 3: Sensitivity and Uncertainty Analysis

- Determination of code uncertainties in RODEX/S-RELAP5
  - basis: assessment of 2 LOFT experiments and CCTF run 79 (only S-RELAP5)
  - method: statistically quantification of differences between calculated and measured PCT
  - uncertainty of measurement covered by uncertainty and bias of code
  - distribution of differences fits a normal distribution
  - results of statistical quantification:
    » blowdown: difference mean = + 33 K
      diff. standard deviation = + 69 K
    » reflood: difference mean = + 130K
      diff. standard deviation = + 127 K
  - blowdown PCT conservative, reflood PCT significant conservative
Comparison of LOFT Blowdown PCTs

Comparison of Combined LOFT and CCTF Run-79 Reflood PCTs
CSAU Element #3: Sensitivity and Uncertainty Analysis

- Determination of uncertainties of statistically treated plant parameters
  - basis: sensitivity studies of 9 parameters (e.g. containment backpressure, initial PRZ level, etc)
  - method: definition of frequency distribution (normal or uniform)
    delta-PCT as a function of plant parameter
  - examples:
    » containment back pressure: normal distribution mean = 0 K sigma = 5 K
    » initial loop flow rate: uniform distribution (-193, +230 kg/s)
      Blowdown delta-PCT = -0.088 K/kg* s
      Reflood delta-PCT = -0.016 K/kg* s
    » rest of 7 parameters treated accordingly
CSAU Element 3: Sensitivity and Uncertainty Analysis

- Determination of uncertainty of fuel related parameters
  - 4 fuel related parameters:
    » core power
    » total peaking factor (max. LHGR)
    » stored energy in fuel (gap)
    » decay heat
  - distribution of frequency of all fuel related parameters: normal
    » core power: mean 100 %, max 106 % (= + 2 sigma), min 97 % (= - 1 sigma)
    » max LHGR: mean 475 W/cm, max 533 W/cm (= + 3 sigma), min 465 W/cm (= - 0,5 sigma)
    » gap: mean 0,095 mm, max 0,11mm (= + 3 sigma), min 0,081mm (= - 3 sigma)
    » decay heat: mean DIN + 0 sigma, max DIN + 2 sigma, min DIN-2 sigma
CSAU Element #3 : Sensitivity and Uncertainty Analysis

• Determination of uncertainty of fuel related parameters (cont)
  – creation of a response surface of PCT in blowdown and reflood

  » definition of design levels : 3 per parameter (mean = 0, max = +1, min = -1)

  » definition of calculation design matrix : 3 level incomplete factorial design results in 25
    runs with code system RODEX/S-RELAP5/COCO

  » stepwise regression analysis for fitting second degree polynom (= response surface):
    PCT = A0 + A1*POWER+ ....+ A5*POWER^2+....+ A9*POWER*MLHGR+......

  » check of fit via 4 additional runs and comparison of response surface to code system
    results
Blowdown Response Surface Versus S-RELAP5 PCT
(including Check Cases)
CSAU Element #3: Sensitivity and Uncertainty Analysis

- Determination of total frequency of distribution of PCT
  - Monte Carlo sampling of:
    » fuel related parameters as input for response surface, calculation of PCT
    » additional uncertainty of response surface fit (delta-PCT)
    » code uncertainty (delta-PCT)
    » plant parameters uncertainty (delta-PCT)

  - PCT = PCT(resp. surf. fuel) + delta-PCT(fit resp. surf.) + delta-PCT(code) + delta-PCT (plant)

  - 50000 trials for blowdown and reflood PCT
Reflood Total PCT(°C) Uncertainty Cumulative Distribution
Reflood Total PCT(°C) Uncertainty Frequency Distribution
CSAU Element #3: Sensitivity and Uncertainty Analysis

- Result for statistical LB-LOCA analysis:

  95% / 95% PCT Blowdown  915 °C
  95% / 95% PCT Reflood    1046 °C

- ECCS design shows sufficient margin against main acceptance criteria of
  PCT < 1200 °C

- Deterministic approach for rest of criteria
APPLICATION OF BEST ESTIMATE METHODS
FOR VVER REACTORS IN HUNGARY

by
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Budapest, Hungary

OECD CSNI SEMINAR ON
BEST ESTIMATE METHODS IN THERMAL-HYDRAULIC SAFETY ANALYSIS
Turkey, 29 June - 1 July, 1998

Abstract

The paper summarizes the problems and experiences in safety analysis faced by a country operating VVER plants. On the example of the large break LOCA case the approach used in conjunction with best estimate codes is described. Future activities are proposed in order to evaluate model uncertainties, which would allow application of a real best estimate methodology with quantitative uncertainty assessment.

1. BACKGROUND

Construction of the first VVER units began in Hungary at the end of the 70-ies. The safety analysis report of the Paks NPP was supplied by the Soviet designers and contained mostly generic VVER-440/213 analyses. With our today's knowledge it can be criticised in different aspects, e.g.: the list of initiating events was not complete, documentation of the results was unsatisfactory, length of the transient times did not assure verification of safe end-conditions, the conservatism applied could not be verified.

The computer codes available at that time in Hungary did not allow independent assessment of plant transients. It was only in the early 80-ies that first thermal-hydraulic system codes, e.g. RELAP4/mod6 became available, which - along with the construction of the first integral-type test facility (PMK) for VVER-440 reactors in Hungary - gave an important impetus to assessment capabilities. Code validation work based on PMK test results started thus increasing confidence in calculations performed for the Paks NPP. The aim of these calculations was first of all to acquire deeper knowledge of plant behaviour and, consequently, the best estimate approach was applied.

As a result, within the country a large amount of experience has been gained in the field of nuclear safety; moreover, knowledge concerning the safety assessment of nuclear power plants has significantly improved world-wide. The systematic utilization of this information was one of the goals of the AGNES project started in 1992.
2. THE AGNES PROJECT

The main objective of the project was the reassessment of the nuclear safety of the Paks NPP and this activity was coupled to the first periodic licence renewal of the plant. The need for thorough reassessment was especially justified because the safety of Soviet designed NPPs could hardly be assessed based solely on the original documentation.

The Technical Scientific Council of the Hungarian Atomic Energy Commission summarized these aims as follows:

- a state of the art report on the reassessment of the nuclear safety of Paks NPP should be prepared,
- the deterministic analyses of design basis accidents and severe accidents as well as probabilistic analyses necessary for the preparation of the report should be carried out,
- the priorities of safety enhancement measures should be determined,
- preparations for elaborating an up-to-date safety report should commence.

The task of the accident analyses was to assess plant safety on the basis of internationally recognized systems of requirements by using generally accepted state-of-the-art tools. All design basis accident (DBA) analyses were repeated in the course of the project, practically providing a new Safety Analyses chapter of a SAR. The analyses included PTS analyses and anticipated transients without scram (ATWS) scenarios. The set of initiating events covered every initiating event considered worldwide as well as specific cases occurring in VVER-type reactors. The most advanced computer codes were applied in the reactor safety studies, and when a system-specific validation was needed to ensure quality of results, this validation was performed. No guidelines for safety analysis were available, so the methodology based on the best estimate codes had to be developed.

Since the plant was licensed on the basis of Soviet norms adapted at that time, special attention was devoted in the AGNES project to defining the acceptance criteria. The acceptance criteria were elaborated by taking into account the regulations in the USA, Germany, France and Finland and they correspond to the European level of expectation.

Later on, the requirements applied in the AGNES project served as a basis for the authority, when stating the licensing requirements for the Periodical Safety Review of the Paks units and also the new legislation, came into force last year, drew much on its basic ideas.
3. THE LARGE BREAK LOCA CASE

The LBLOCA accident proved to be one of the most limiting cases in the AGNES project. While transients and accidents analysed for overpressure, DNB, overcooling and reactivity insertion fulfilled the acceptance criteria with fairly large margin and in small break LOCA accidents even the minimum configuration of ECCS assured adequate core cooling, this latter was not true for LBLOCA.

3.1 AGNES LBLOCA findings

As mentioned before, TH code applications in Hungary of the 80-ies focused on SBLOCA and transients. RELAP4/mod6 was used for LBLOCA analysis within IAEA projects, but by the time RELAP5 became available to us, USNRC did not support it anymore as a LBLOCA tool. Hungary has never subscribed with NRC to the TRAC code, so when for the AGNES project LBLOCA analyses had to be performed, IVO’s suggestion was followed and the calculations were executed under a bilateral agreement with GRS, using the 1.1 Cycle A version of the ATHLET code.

The way, the ATHLET code was applied, reflected the state-of-the-art of most Western countries in the early 90-ies. The code was developed and validated by GRS on a series of separate effects and integral tests [2]. Although all these tests concerned Western type PWRs, it was taken for granted that the code can be used with confidence for VVER type plants as well. (It is true that no other code in the world was formally validated for VVERs and VVER-specific experimental data for LBLOCA were - and still are - practically non-existing.) However, the advantage of the code was that - following the political changes in Europe - it has been extensively used for the analysis of VVER plants in and outside Germany and so large experience in modelling of this type of NPPs has been acquired. As a consequence, it was impossible to perform an uncertainty analysis in the sense it is required e.g. by the USNRC Regulatory Guide 1.157 [3]. In order to assure that the calculated key parameter values bound the uncertainties resulting from code model imperfections the following conservative steps were adopted:

- The worst possible break size was found from a spectrum of large breaks to be the double ended guillotine break of the cold leg.
- Nodalisation (especially in the downcomer, core and upper plenum regions) was carefully selected in order to pessimistically model phenomena for which the code could not be validated, e.g. hydroaccumulator water breakthrough from UP, bypass of ECC water directly injected to the DC by HA and LPIS.
- The initial operating conditions of the plant were set at their bounding limits, the core power being 104%.
- Highly pessimistic moderator density reactivity coefficients were used.
- Beginning of life power peaking factors with end of life decay heat were selected. Three hot rods were modelled in the hot assembly with BOL, MOL and EOL axial power distributions, respectively, and the one with the highest peak cladding temperature (PCT) was compared to the criteria.
- Fuel gap conductivity was entered with pessimistically low value.
• ECCS water temperature was pessimistically high, coupled with minimum flow rate from the active systems.
• Single failures (including SF of passive systems as well) were evaluated by parametric runs, the most adverse one, non-availability of one HA injecting to the UP was retained.
• The water injected by the LPIS into the broken loop was supposed completely lost.
• The steam generator secondary side was isolated after turbine trip that maximises reverse heat transfer and results in increased steam binding.
• Parametric studies revealed that the case with loss of off-site power was more limiting: this assumption was retained.

The cumulation of all these pessimistic assumptions led to a PCT of 940 and 1000 °C, for a downward skewed and a cosine axial distribution, respectively, as shown in Fig. 1. This represented a comfortable margin with respect to the acceptance criterion of 1200 °C.

3.2 Changes in ECCS start-up logic

As a result of the AGNES project the safety enhancement proposals for the Paks NPP have been reviewed and prioritised. Changing the ECCS start-up logic figured among the high-ranked issues. It is a common feature of the original design of VVER plants that upon arrival of an ECCS signal the internal grid is automatically isolated from the net, even if off-site power is available, and the different consumers needed for emergency core cooling are gradually supplied by the diesel generators. This means an “artificial” blackout of all AC power to the plant for the period of the diesel generators reaching the correct frequency, the duration ranging between 10 and 30 seconds. The probable reason of this approach was the highly unreliable external grid system at the time of design of VVERs in the 1970-ies, where the accidental loss of a unit could lead with a high probability to the collapse of the grid and the consequences could be handled in a more systematic way by the “artificial” blackout. Nowadays, when the Hungarian grid can withstand with high probability the loss of one of its nuclear units, PSA studies have shown the adverse effects of this start-up logic. As a consequence, a change was proposed, which consisted in supplying the ECCS from the grid - if this is available - and only making use of the diesel power, if the AC power is effectively lost.

One of the safety analyses affected by the proposed change was the LBLOCA case. The main impact is that - in contrast to earlier studies - the loss of the grid has to be supposed any time during the transient. Parametric studies performed have shown that the most adverse timing of the loss of AC power to the plant with respect to PCT is the moment, when the maximum cladding temperature is reached. If the ECCS (especially the low pressure pumps) loose their electric supply until they are backed up from the diesel generators, this means a delay in the rewetting process and results in further increase of the cladding temperature. The evolution of the hot rod cladding temperature is shown for this case in Fig. 2: as a result of the modification the maximum temperature has increased by about 100 °C. It should be mentioned that in the LBLOCA case there is practically no positive influence due to the immediate start-up of ECCS systems within the modified logic, since a considerable
part of the ECCS water anyway bypasses the core to the break during the blowdown period and, on the other hand, the shut-off head of the LPIS pumps does not allow injection before the hydroaccumulators are empty, thus there is practically no change in the blowdown period, whichever start-up logic is applied.

In view of the largely unknown uncertainties related to the application of the ATHLET code the margin of 100 °C was judged insufficient and AEKI, responsible for the analysis, did not support the modification for LBLOCA. The utility took the decision to speed up diesel start-up, if this significantly influenced PCT. The analysis performed with a start-up time of 10 s instead of 24 s yielded a reduction of the PCT just below 1050 °C and the utility decided to submit the modification with these figures to the authority for approval. Questions raised during the review process obliged AEKI to reconsider the conservative assumptions applied in the analysis.

3.3 Review of conservatisms

The review of the applied conservatisms focused on items, which involved unrealistically pessimistic or contradicting assumptions, but did not touch to the basic approach, i.e. that the non-quantified uncertainties related to the application of the best-estimate code ATHLET should be counterbalanced by a sufficient number of pessimistic assumptions. After careful analysis of the input data the following corrections have been performed:

- For the AGNES project the reactivity feedback coefficient of moderator density was defined very pessimistically with a conservative value for the nominal state that was supposed to be valid over the whole density range. Development of the 3-D reactor physics code KARATE [4] allowed the definition of the moderator density coefficient over the whole applicable density range taking into account the model uncertainties, which resulted in much less restrictive feedback values at low densities.
- It was considered unnecessary to apply conflicting axial power distributions for the the modelled hot and average assemblies and the hottest rod.
- Unrealistically low pellet-cladding gap conductance value was applied by error for the middle-of-cycle (MOC) state in the AGNES project: this has been corrected. Since there is no gap conductance data supplied by the fuel vendor, pessimistic data from the literature had to be used. It is planned to derive more realistic data based on the research to be performed in AEKI’s Reactor Material Laboratory.
- AGNES LBLOCA analyses only took into account the engineering factor for the hot rod and not for the hottest assembly to which the hot rod was linked hydraulically. It was decided that the analyses should - as an additional pessimism - consider the engineering factor for the hot assembly as well.

The AGNES results have been repeated with the above changes. The PCTs during the blowdown and the reflood phase are compared in Table 1. The overall effect of the modifications resulted in a reduction of the PCT by 140 °C, the main contributor being the void coefficient, the gap conductance coming in second place. The added pessimism of the engineering factor resulted in an increase of about 50 °C.
Table 1. AGNES PCTs (°C) with original and revised conservatisms

<table>
<thead>
<tr>
<th>Calculation</th>
<th>Blowdown PCT</th>
<th>Reflood PCT</th>
</tr>
</thead>
<tbody>
<tr>
<td>AGNES/original</td>
<td>875</td>
<td>940</td>
</tr>
<tr>
<td>AGNES/revised</td>
<td>840</td>
<td>800</td>
</tr>
</tbody>
</table>

From the results it can be concluded that by suppressing a few highly pessimistic assumptions the PCT could be decreased by 140 °C. It is to be noted that in the revised calculation the PCT occurs no more in the reflood phase but during blowdown, which is the result of the lower cladding temperatures: the contribution of zirconium oxidation is negligible in this range.

4. FUTURE PERSPECTIVE

Obviously, the method described above for the LBLOCA analysis differs considerably from either the USNRC and the IAEA guidelines [5], in that both rely on the quantification of the uncertainties, or, if this proves to be difficult, on the inclusion of bias bounding specific uncertainties. Here, instead, a large number of conservatisms was cumulated, but it would be very difficult to demonstrate mathematically that the resulting key parameter values bound all the uncertainties. That is why sufficient margin with respect to the acceptance criteria has to be maintained.

In order to correctly apply the best-estimate methodology the following steps should be taken:
- The most important phenomena affecting the key parameters should be identified.
- Models describing the above phenomena should be validated by test results.
- A complete uncertainty analysis should be performed.

A number of activities has already been started, the problems encountered and the experiences gained are summarised below.

Identification and ranking of important phenomena
This can be done on the basis of previous work performed for PWRs, e.g. based on the Phenomena Identification and Ranking Table of the CSAU for LBLOCA [6], while differences between PWRs and VVERs can be evaluated by the help of the recently developed OECD VVER-specific validation matrix [7]. In fact, in a recent PHARE project the following table was proposed for LBLOCA analysis, based on the above mentioned references. Only the high ranked parameters of [6] were included into the table.
Table 2. High ranked phenomena affecting key parameters in LBLOCA

<table>
<thead>
<tr>
<th>Important phenomena</th>
<th>Blowdown</th>
<th>Refill</th>
<th>Reflood</th>
<th>VVER specific</th>
</tr>
</thead>
<tbody>
<tr>
<td>Fuel stored energy</td>
<td></td>
<td></td>
<td>X</td>
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</tr>
<tr>
<td>Fuel oxidation</td>
<td></td>
<td>X</td>
<td></td>
<td></td>
</tr>
<tr>
<td>Decay heat</td>
<td></td>
<td></td>
<td></td>
<td>X</td>
</tr>
<tr>
<td>Gap conductance</td>
<td></td>
<td></td>
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<td></td>
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<tr>
<td>Post-CHF heat tr.</td>
<td></td>
<td></td>
<td></td>
<td>X</td>
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<tr>
<td>Rewet</td>
<td></td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>Reflood heat tr.</td>
<td></td>
<td>X</td>
<td>X</td>
<td></td>
</tr>
<tr>
<td>Core void distribution</td>
<td></td>
<td></td>
<td></td>
<td>X</td>
</tr>
<tr>
<td>Entrainment, deentr. in UP</td>
<td></td>
<td></td>
<td></td>
<td>X</td>
</tr>
<tr>
<td>Steam binding in SG</td>
<td></td>
<td></td>
<td></td>
<td>X</td>
</tr>
<tr>
<td>2-phase pump behaviour</td>
<td></td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>Pump losses</td>
<td></td>
<td></td>
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<tr>
<td>Condensation by SI</td>
<td></td>
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<td>X</td>
</tr>
<tr>
<td>Entrainment, deentr. in DC</td>
<td></td>
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<td></td>
<td></td>
</tr>
<tr>
<td>Break flow</td>
<td></td>
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</tr>
<tr>
<td>Loop oscillations</td>
<td></td>
<td></td>
<td></td>
<td>X</td>
</tr>
</tbody>
</table>

Although also in this PHARE project the ATHLET code was used for the analysis, a more detailed assessment of code uncertainties was beyond its scope: for each of the items in Table 2 only the references to test results could be collected, against which the ATHLET code was validated.

Quantitative assessment of uncertainties
As a further step the quantitative assessment of the uncertainties based on the above mentioned validation results could be performed. These would, however, be restricted to phenomena, which are not VVER-specific. As far as VVER-specific test data for LBLOCA are concerned the problem is faced that these are practically non-existing - with the exception of some small scale reflood data. So - if uncertainty analysis for VVERs should be developed - reliable experimental data needs to be produced first.

ATHLET uncertainty evaluations
AEKI has started a project - under its bilateral agreement with GRS - for the quantification of ATHLET code uncertainties for VVER systems. As a first step the GRS uncertainty method will be applied to a medium size LOCA experiment performed last year at the PMK facility. The test models the guillotine break of the pressuriser surge line. It is hoped that some of the questions raised in Table 2 can be answered once results of the project will be available.

OECD CSNI LBLOCA activities
The PWG2 has just issued the results of a several year activity aiming at the comparison of different uncertainty methods based on a SBLOCA LSTF experiment [8]. Discussion within PWG2 and its Task Group on Thermal-Hydraulic Applications on a possible follow-up activity concluded that the analysis of uncertainties in
LBLOCA is of great importance and therefore application of uncertainty methods to the analysis of a LBLOCA test was proposed. This activity will be coupled to revisiting of an old International Standard Problem on LBLOCA.

5. CONCLUSIONS

The level of accident analyses included in the Safety Analysis Report of the Paks NPP obliged Hungary to deploy an important effort. Starting from the 1980-ies, internationally recognised best estimate computer codes became available to the country and, in parallel, substantial work was invested into the qualification of these codes by VVER-specific test results, first of all based on the PMK facility.

A review of the safety of the Paks plant in the AGNES project has shown that VVERs have large margins. This allows application of best estimate codes in combination with a bounding approach, where most uncertainties are bounded by conservative assumptions. It was shown that the LBLOCA case is one of the accidents where this approach may lead to limitation and therefore a real best estimate methodology should be followed. The main difficulty is the lack of VVER-specific large break test data.

In order to arrive at a “better estimate” LBLOCA analysis, the following steps are proposed:

- identification of the most limiting phenomena;
- execution of the necessary tests to cover these phenomena as a follow-up of the establishment of the OECD VVER-specific code validation matrix;
- quantification of the most important uncertainties based on the test results.
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Fig. 1 Maximum cladding temperatures

Fig. 2 Maximum cladding temperatures
BEST-ESTIMATE METHODS IN CANDU REACTOR LOCA ANALYSIS

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ABSTRACT

The Best Estimate (BE) approach, used within the Canadian Nuclear Industry, is presented in this paper. BE codes used for analysis of the CANDU® reactor are being developed primarily by Atomic Energy of Canada Limited (AECL) and Ontario Hydro Nuclear (OHN). The CATHENA and TUF two-fluid thermalhydraulic computer codes are first described. Analysis requirements for the CANDU system are outlined. The procedures for coupling the thermalhydraulics to the plant control, reactor physics, fuel element and fuel channel codes are then given. The importance of validation of the codes with the associated uncertainty analysis is stressed. Finally, an example of a coupled calculation is given.

The Utilization of Best Estimate Methodology in Reactor Safety Analysis

OECD/CSNI Seminar
Ankara, Turkey
1998 June 29 - July 1

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1. INTRODUCTION

In the early years of safety analysis, conservative bounding analyses were used for plant licensing evaluations. With improved understanding of some of the underlying physics and thermal-hydraulic phenomena, and the desire to reflect this knowledge in advanced two-fluid thermal-hydraulics codes, has come a move toward best estimate safety analyses. This, in turn, has resulted in the need to more formally and rigorously validate these computer codes and assess the results obtained with formal uncertainty analyses. These are current topics of significant investigation in Canada, as well as internationally, as reflected in the work of the OECD/CSNI/NEA Principal Working Group 2. Recently a CSNI status summary on the utilization of best-estimate (BE) methodology in safety analysis and licensing has been released (Reference 1).

The unique design features of the CANDU reactors and the intrinsic safety related features distinguish it from other types of reactors. A CANDU reactor is a pressure tube, heavy-water-moderated, high pressure heavy-water-cooled reactor consisting of two figure-of-eight loops. The fuel channels consist of two concentric tubes (pressure tube and calandria tube) with a space in between filled with an inert gas. The pressure tubes (390 to 480, depending on the design) are horizontal and are rigidly jointed to end-fittings which are firmly supported by the end shields. This arrangement allows fueling of the reactor during normal operation. The calandria tubes, which are roll expanded at both ends into the calandria side tubesheets, separate the cold moderator in the calandria vessel from the hot pressure tubes.

System thermalhydraulics analysis codes are key computational tools in the nuclear industry and are designed for application in safety and licensing analyses. To provide BE calculations of certain postulated upset and accident scenarios, coupling with other codes (e.g. reactor kinetics) is required.

This paper first provides a brief summary of the features a CANDU reactor. Descriptions of the CATHENA and TUF two-fluid thermalhydraulic codes are given followed by analysis requirements for CANDU safety and licensing. Coupled calculations between the thermalhydraulics and other calculational modules are discussed. The role of validation and uncertainty analysis on BE methods is briefly reviewed. Finally an example of a coupled calculation is given.

2. THE CANDU REACTOR

Figure 1 shows a simplified schematic of the major process systems common to all CANDU designs.
Moderator system

A noted feature of the CANDU reactor is the large, horizontal, cylindrical calandria vessel containing heavy water which serves as the neutron moderator. Several hundred Zircaloy tubes, called calandria tubes, are arranged within and parallel to the cylindrical shell. They are fastened with leak-tight joints to tubesheets at the ends of the calandria. The heavy water moderator is operated at essentially atmospheric pressure and the temperature is maintained well below the boiling temperature of water at the operating pressure.

Reactor Heat Transport System and Fuel Channels

The reactor heat transport system is arranged in either one or two figure-of-eight loops in which large-diameter piping connects major components, such as the steam generators and pumps, to the reactor header pipes. Approximately 100 distributed, small-diameter feeder pipes are connected from these large-diameter headers to the end fittings of the horizontal fuel channels. The end fittings are special components designed to accommodate on-power refuelling. Figure 2 shows a simplified schematic of a typical two-loop CANDU heat transport system design.

The fuel channel assemblies consist of Zr-2.5% Nb alloy pressure tubes connected by rolled joints to the body of the end fittings at either end of the reactor. The pressure tubes are contained within thin-wall Zircaloy calandria tubes, and the annulus between the two tubes contains slowly circulating carbon dioxide gas, which helps to insulate the hot pressure tube from the cold moderator fluid surrounding the calandria tube. The two tubes are separated by a series of coiled Zircaloy springs, called garter springs, placed at a number of locations along the pressure tube.

Each pressure tube contains either 12 or 13 fuel bundles, depending on the reactor design. These fuel bundles (see Figure 3) are approximately 0.5 m long and contain 28 or 37 fuel elements configured in arrays of concentric rings of elements welded to the end plates. The fuel elements are made of sintered pellets of natural uranium dioxide contained within thin-walled Zircaloy sheathing tubes designed to collapse onto the pellets at operating pressures. As mentioned earlier, a unique feature of CANDU reactors is on-power fuelling, where irradiated bundles are discharged at one end of a channel and new bundles are inserted at the other end.

Safety Systems

Safety systems include the reactor shutdown systems, the emergency core cooling system and the containment.

Two independent reactor shutdown systems are provided. Additional negative reactivity can enter the reactor core through tubes installed into the reactor from the top and sides. Shutdown System 1 employs a large number of distributed shut-off rod units that can be driven rapidly from above into the core by spring-assisted gravity action. Shutdown System 2 relies on the gas-driven injection of heavy water containing a dissolved absorber ("poison") into the moderator.
CANDU reactors, in common with other reactors, are fitted with emergency core cooling systems and containment systems. The emergency cooling systems are designed to operate over the complete range of postulated loss-of-coolant accidents, and can operate at high and low pressures.

Containment systems fitted to CANDU reactors vary from multi-unit systems with water spray and vacuum building pressure suppression to single-unit systems that rely on the reinforced concrete containment vessel without pressure suppression.

3. CATHENA

CATHENA, developed by AECL, has evolved with the objective of providing a high degree of flexibility in modeling thermalhydraulic systems. Although developed primarily for the analysis of CANDU nuclear reactors, the code has been successfully applied in the analysis and design of experimental test programs. CATHENA is also being used in support of the design, safety and licensing of research reactors developed by AECL (e.g., MAPLE-X10, Reference 2). A comprehensive description of the code can be found in Reference 3.

Thermalhydraulic Model

The CATHENA code uses a non-equilibrium, two-fluid thermalhydraulic model to describe fluid flow. Conservation equations for mass, momentum and energy are solved for each phase (liquid and vapour), resulting in a 6-equation model. Also, up to four noncondensible gases may be represented as part of the vapour phase, yielding a 7- to 10-equation model. Interphase mass, momentum and energy transfer are flow-regime-dependent, and are calculated using constitutive relationships obtained from the literature or are derived from single-effects experiments.

Noncondensable gas properties are available in CATHENA for H₂, He, N₂, Ar, CO₂ and air. The thermodynamic properties for the noncondensable gases are assumed to follow the ideal gas law.

The numerical solution technique used to solve the conservation equations is a one-step, staggered-mesh, semi-implicit, finite-difference method. The dependent variables defining the state of a node or cell are pressure, void fraction, and phase enthalpies. If noncondensable gas(es) are present, the noncondensable fractions are also dependent variables. For connections between nodes (called links), the dependent variables are the velocities of the gas and liquid phases. Conservation of mass is achieved using a truncation error correction technique similar to that used in RELAP5/MOD2 (Reference 4).

A one-step finite-difference numerical solution scheme has been adopted that is not transit-time-limited. The resulting set of equations is not reduced to a pressure- or flow-field approach. A time-step controller implemented in CATHENA automatically selects the next time step at each finite-difference time step. This is accomplished by monitoring changes in the dependent variables, the selected derived variables, and the truncation error. If the maximum change is
below a prescribed value, the time step is increased; if the change is above a maximum prescribed value, it is decreased. The user may alter the default selection criteria through input data and thus check the temporal convergence of a given simulation.

Heat Transfer

Heat transfer from metal surfaces is handled by an extensive wall-heat-transfer package, GENHTP (GENeralized Heat Transfer Package). A set of flow-regime-dependent constitutive relations specifies the energy transfer between the fluid and the pipe wall and/or the fuel element surfaces. A variational finite-element method is used to model the heat transfer by conduction within the piping and fuel in the radial direction, and the heat transfer can also be modeled in the circumferential direction. The radiative heat transfer and the zirconium/steam reaction rates can also be calculated. The ability to calculate the heat transfer from individual groups of pins in a fuel bundle subjected to stratified flow is built into this package. Under these conditions, the top pins in a bundle are exposed to vapour, while the bottom pins are exposed to liquid.

The GENHTP model also allows for the calculation of pressure tube (and calandria tube) strain, caused by pressure tube heat-up in a pressurized channel. Under certain postulated conditions, the pressure tube can “balloon” into contact with the calandria tube. When this occurs, heat will be transferred from the channel, through the pressure tube and calandria tube to the D₂O moderator. CATHENA is able to model all these processes, including the contact conductance between the pressure tube and calandria tube.

CANDU fuel is usually modeled by a user-specified UO₂ region, gap, and fuel sheath region. Temperature-dependent properties, available within GENHTP, are generally used for the UO₂ and sheath. Although a constant gap conductance is usually used for LOCA analysis, a user-specified time dependent value may be applied.

Component Models

Component models that describe the behavior of pumps, valves, pressurizers, steam separators, and discharge through breaks are available to complete the idealizations of the reactor systems.

The Generalized Tank Model (GTM), a two-region, two-fluid, non-equilibrium thermalhydraulic model is provided to model tanks, allowing a variable cross-sectional area. The upper and lower tank regions are modeled as independent volumes which are allowed to exchange heat and mass through mechanisms such as condensate fall, bubble rise or inter-region condensation.

Modeling Control Systems

Control systems may be modeled using a simulation-like language within CATHENA. Included within this system is the ability to use FORTRAN-like statements. The majority of variables calculated by CATHENA can be accessed, and used in these calculations. The results of these calculations can then be used to “control” models such as valve (opening fraction), reactor kinetics (reactivity insertion), etc.
In some cases, complex reactor control systems are already available in FORTRAN code. For these cases it is more efficient to use the existing control program, and "couple" it with CATHENA. CATHENA provides the required information to the Controller Program, and the Controller Program provides the required valve opening fractions, etc to CATHENA. This approach has been used to couple CATHENA and LEPCON (the plant control routines for the Point Lepreau CANDU 6 reactor, Reference 5).

**Steady State Initialization**

The CATHENA code obtains a steady-state solution by simulating a transient with arbitrary but reasonable initial conditions. Boundary conditions are specified representing the steady state condition. The steady-state solution is obtained when flow and heat transfer rates do not vary in time. The user can use his/her experience in determining the length of the transient to reach time-invariant conditions, or this may be performed automatically by the code.

4. **TUF**

The TUF code was developed at Ontario Hydro, for the thermalhydraulic analysis of Ontario Hydro's operating reactors. It consists of two separate programs, one which provides a steady-state solution and the other a transient solution that is initialized to the steady state solution.

**Thermalhydraulic Model**

The two-fluid equations are reduced to a set of flow rate equations, which has a matrix size (LxL) where L is the total number of links (excluding boundary links) and run-down pumps. In the steady state program, the thermalhydraulic variables are pressure, specific enthalpies for mixture and vapour, quality, mixture flow rate and slip velocity. In the transient program, the variables are mixture mass, vapour mass, mixture internal energy, vapour internal energy, mixture flow rate and slip velocity. Currently, there are two numerical methods available in the code: the one-step semi-implicit method and the simple two-step implicit method. The simple two-step implicit method is a predictor-corrector type technique. In the first step, the mass and energy conservation equations are solved explicitly. These solutions are only used to update the link properties used in the transport terms, where the variables associated with pressure wave (pressure and mixture density) are not updated. In the second step, the implicit method is applied to obtain the final solutions. To verify the numerical method implemented in the code, the JUICE 1976 standard problems were simulated (Reference 6) and compared with the so-called exact numerical solutions obtained from the MECA code (Reference 7). Comparison of initial pressure oscillation in the problem of instantaneous heat addition (Standard Problem 1) of flowing sub-cooled water at high pressure in a vertical pipe indicates that the TUF solution displays a small amount of numerical damping in addition to the frictional damping associated with the problem. Nevertheless, numerical diffusion inherent in TUF is much less than in FIREBIRD, RAMA, and RELAP-UK codes as shown in (Reference 8).
Component Models

TUF contains modules dealing with thermalhydraulics (one-fluid, drift-flux and two-fluid), reactor physics (point kinetics or external coupling with other reactor physics codes), heat conduction (pipe wall, heat exchangers, pressure-calandria tubes and fuel pins), system components (pumps, valves, boilers, pressurizer, bleed condenser, turbine and accumulator), and special models (discharge model, level swell, bundle movement, pressure tube strain model and metal-water reaction). Special attention has been given to the accurate representation of plant components.

Modeling Control Systems

The TUF code must provide an accurate simulation of automatic controller actions and operator interventions in any plant operational analysis. The reactor controllers simulate the following control systems: overall unit control, reactor regulating system, steam generator pressure and level controls, heat transport (HT) system pressure and inventory controls, bleed condenser pressure and level controls and safety systems. In the TUF code, although the control routines differ from one station to another, the code has been designed so that the control elements and the locations at which the input variables are measured are easily identified by means of location codes assigned in the input data for all CANDU reactors. Also, users can assign different control states for each major control function through the input data. For example, the overall unit control can be assigned either in normal or manual mode. The steam generator pressure control can be either in warm-up, cool-down or normal hold mode.

An overview of the modules implemented in the TUF code and their role in the operational support and licensing safety analyses can be found in (Reference 9).

Steady State Initialization

Many of the existing reactor system codes do not provide a steady state program. However, in the TUF steady state program, the equations dealing with thermal-hydraulic variables, nodal heat flux, heat exchanger film resistance and valve position (or special link resistance) for a control system are solved. The set of simultaneous non-linear equations is solved by the Newton-Raphson iteration method. To match the steady state solutions with normal operating conditions, different control flags are used in the input data. These flags are used to define the degrees of freedom for the steady state simulation, particularly when the control systems are involved.

5. ANALYSIS REQUIREMENTS

The primary application of system thermalhydraulic codes are in a) safety analyses of accident and plant upset transients, b) operational support which includes simulating plant transients and assessing the behaviour of various systems, and c) design assist analysis for new designs and modifications to existing designs. Because of the high cost of implementing Quality Assurance
(QA) procedures for a given code, as well as the required validation, it is advantageous to have one thermalhydraulic code for a wide range of analysis requirements.

Safety analyses require a wide range of detailed phenomenological modeling and an ability to represent behaviour in events with widely varying time scales. For example, both short duration (i.e., waterhammer) as well as long duration transient events associated with specific accident initiating failures have to be accommodated. In some accidents, such as large LOCA, many interacting phenomena occur, requiring interaction between all the major disciplines, including reactor physics, thermalhydraulics, fuel and fission product behaviour, fuel channel thermal-mechanical response, containment response and moderator system behaviour. Other accident events, such as small LOCA and secondary side failures can extend over relatively short to very long time periods. Additionally, accident events invariably involve physical phenomena whose modeling represents current state-of-the-art.

Operational support analysis places a very high demand on the ability to represent the plant systems and components with high fidelity. Since, by definition, operational support tends to be best estimate analysis of the plant behaviour, it very often requires detailed modeling of specific plant components, such as steam generators, pumps, pressurizer, bleed condenser and various valve types, as well as the various plant controls. This requirement has been a strong driver in the development of the SOPHT (see Appendix A) and TUF codes and is responsible for the control algorithms being simulated in dedicated software routines for each of the Ontario Hydro stations. Additionally, the requirement for upward compatibility in migrating from the SOPHT to the TUF code dictated that the investment in detailed modeling of the control algorithms be maintained by ensuring that the routines from SOPHT be directly transportable into TUF.

Design assist analysis is an important application for both new designs and modification to existing designs. As in the case of operational support analysis this application area requires attention to the detailed representation and modeling of interacting plant systems, major plant components and controllers.

6. **COUPLED CALCULATIONS**

As discussed previously, for certain analysis requirements, calculations are required that are coupled to other codes. In Reference 10, the interfacing of the thermalhydraulic model to other code modules is discussed. Seven overall modules are identified here:

1. Thermalhydraulic Module,
2. Heat Conduction and Heat Transfer Module,
3. System Components Module,
4. CANDU Controllers Module,
5. Reactor Physics Module,
6. Fuel Channel Module, and
A given code may contain more than one module, and in this case the modules are usually "tightly" coupled. The TUF code contains Modules 1 through 4, while the CATHENA code contains Modules 1, 2, 3, and 6.

The main computational tools used in the safety analysis are described in Appendix A.

The interfacing of the modules contained in separate codes is described next.

6.1 Thermalhydraulics and Reactor Physics

An interface between thermalhydraulics and reactor physics codes is normally used in the analysis of a large LOCA in the safety analysis. It would be a very expensive model to include all the reactor channels in the thermalhydraulic circuit of CANDU reactors. Normally, the averaged channel approach is adopted. For example, each reactor core pass can be simulated by one region (or zone) with averaged channels or several regions each having its own averaged channels. Considerable effort and thought goes into selecting these regions, taking into account such physical parameters as the elevation and regional power of groups of channels.

Approach at AECL

CATHENA is coupled with the RFSP code (Reference 11), and the coupling methodology is shown in Figure 4.

The CATHENA model of the core made up of 380 fuel channels in a two-loop CANDU 6 reactor typically comprises eight channel groups. The core-pass downstream of the break is modeled with 5 channel groups divided radially and by different elevations. The 95 channels in the return pass of the broken loop are represented by one group. The passes in the intact loop are represented by 2 channel groups based on the headers to which they are connected. The core region in each channel group in the critical pass is represented by 12 thermalhydraulic nodes chosen to contain one axial bundle plane each. This is done to ensure sufficient accuracy in the prediction of coolant density in the core region.

For the physics calculations in RFSP, the transient analysis involves shutdown system 1 (SDS1). Therefore models of the shut-off rods are included.

Steady state

To generate a steady state from which to initiate the break, the steps below are followed.

- The initial steady-state thermalhydraulic run is carried out at 100% power with the CATHENA input file containing the geometric input data for the reactor. The desired inlet-header temperature is obtained by adjusting the outlet-header enthalpy in this input file.
- The initial steady-state flux distribution is determined by first guessing an RFSP flux distribution and using a typical average coolant density, coolant temperature and fuel
temperature at 12 average bundles in a channel group in each core pass from the CATHENA steady-state calculation. The results are passed to CATHENA which performs another steady-state calculation. RFSP is executed again with new input from CATHENA. The process is repeated until the new flux distribution converges to the previous calculation.

**Transient calculation**

At each time step, the transient calculation proceeds as follows:

- RFSP calculates flux and power distributions including fission product decay power over discrete “flux-shape” time steps.
- RFSP sums up bundle powers for each of the channel groups and passes this data to CATHENA.
- CATHENA uses this information to compute coolant densities, fuel temperatures and coolant temperatures which are passed to RFSP for the next time step.
- RFSP computes the times at which SDS detectors reach their set-points based on detector fluxes, obtained by 3-dimensional interpolation of the flux distribution. This allows the determination of SDS actuation times.

The coupling is done using the power generation, coolant density, coolant temperature and fuel temperature in each CATHENA node. The neutron flux shape is calculated in RFSP at time intervals Δτ. These time intervals are not necessarily constant through the transient. For a given time interval Δτ, RFSP uses the core coolant densities, coolant temperatures and fuel temperatures produced by CATHENA and determines the 3-dimensional power generation in the core at the end of the interval. CATHENA uses this power distribution to calculate the core node coolant densities, coolant temperatures and fuel temperatures for the next time interval.

In summary, in each “flux-shape” time interval Δτ the CATHENA calculation is done first, followed by the RFSP calculation. Values of Δτ, typically in the range of 0.05 - 0.10 second are used until the shutoff rods are fully inserted. These small values ensure good convergence in the time behaviour of the RFSP solution. Following shutoff rod complete insertion, the flux-shape changes are small and the larger values of Δτ (~1.0 second) are appropriate. Within the time interval Δτ, the CATHENA calculation uses its own time step of the order of 1 millisecond and assumes a constant core power distribution.

**Approach at Ontario Hydro**

The reactor system code TUF can be operated in two modes for the reactor core power calculation: self initiating or iteratively with 3D reactor physics codes. In the self initiating mode, the point kinetics model is used. In the iterative mode, a procedure to couple the thermalhydraulics and reactor physics codes is applied. In the iterative procedure, the reactor physics code SMOKIN (Reference 12) performs a rigorous three-dimensional neutron calculation using the results of coolant density and fuel temperature at each axial node of the various regions obtained from TUF. The regional power transients, as computed by SMOKIN, are then fed back to TUF to obtain a
subsequent set of channel conditions for the reactor physics codes. The use of calculated regional power transients from SMOKIN replace the point kinetics calculations in the TUF code.

To illustrate the actual interaction between TUF and SMOKIN, the approach used in the Bruce station safety analysis for a large LOCA at OHN is described here. The TUF model groups the 480 channels into a number of thermallydraulically similar regions maintaining such details as inner and outer flow zone boundaries and the elevation of channels corresponding to broken and unbroken passes in each region. Two notable conservative assumptions made in the simulation are: (1) an initial bottom-to-top flux tilt of 20 percent and (2) not crediting the three most effective shut-off rods when the shut-down system is activated. The tilted initial power distribution is defined by SMOKIN. This power distribution and the channel grouping definitions are used to produce the initial tilted powers in the TUF model. The channel groups one to five represent the outer flow zone whereas the channel groups six to ten represent the inner flow zone. The calculation procedure is as follows:

1. First, the main conservative assumptions used in SMOKIN and TUF are established. These assumptions take into account the uncertainty in the parameters of the reactor kinetics model and the power measurement.

2. The self-initiating mode for reactor power (using the point kinetics model) in TUF is used to simulate this particular event up to 5 transient seconds. Reasonable reactivity tables for the fuel temperature and coolant void are estimated from the previous SMOKIN runs. Local power in each region and node is estimated by multiplying the core average power with some appropriate scaling (or peaking) factors. The reactivity control mechanisms in the reactor regulating system calculate the total reactivity including the action of shut-off rods. The averaged reactor power is calculated from the point kinetics model. This mode of simulation can provide a first degree of approximation for coolant density and fuel temperature in all the core regions.

3. Using the thermalhydraulics data from TUF, SMOKIN simulates the response (up to 5 seconds) of the neutron overpower system (the shut-down system) and calculates the core reactivity which depends upon the distribution of flux and precursor population as a function of time. Then, SMOKIN supplies the power transients for all the core regions modeled in TUF.

4. By-passing the point kinetics routine, TUF directly uses these region power transients as calculated by SMOKIN in modeling the response of the reactivity devices in the subsequent TUF simulation. TUF re-calculates the core coolant density and fuel temperature variations as function of time.

5. Steps (3) and (4) are repeated until the process converges to a consistent set of thermal-hydraulic, reactivity and fuel power transients. This is achieved when the difference between consecutive SMOKIN results is less than 1 percent. The converged power transient in each region obtained from SMOKIN including the long term decayed power is utilized in the TUF simulation. The SMOKIN results of detailed channel power distributions for all core
channels including the hot channel power are transferred to the fuel channel codes CHAN, SMARTT, ELOCA and FACTAR codes for detailed channel calculations (channel thermal/mechanical behaviour, amount of fission products release, channel integrity, see Appendix A).

The TUF and SMOKIN iteration usually converges quickly after the first pass. This process of iteration, however, has to be carried out for each transient case. It results in a vast amount of data being transferred from TUF to SMOKIN and vice versa. Consideration is being given to the direct coupling of these two codes (TUF and SMOKIN) at OHN since it will not only simplify the calculation procedure but it will result in considerable saving in time when performing similar calculations for scoping analysis.

6.2 Thermalhydraulics and CANDU Controllers Module

The approach, at OH, has been to include the Controllers Module within the TUF code. A unique TUF version is created for each of OH nuclear plants. Thus there is a very close coupling.

The CATHENA approach has been to exchange information, at every timestep, between the Thermalhydraulics Module, and the separate Controllers Module through an “interface” (see Reference 5), or through user input.

6.3 Thermalhydraulics and Fuel Element and Fuel Channel Modules

During the early fuel heatup in a large LOCA, a large temperature difference exists between the fuel centre-line and sheath. A detailed fuel pin model to predict the fuel temperature profile is needed. In the late fuel heatup period due to a degraded core cooling, this temperature difference becomes small and a detailed fuel pin model is not necessary.

Fuel Element Module

For the Fuel Element Module, the main physical parameters are fuel thermal properties (functions of fuel burnup rate and fuel temperature), gap heat transfer coefficient (including fuel swelling effect), flux depression, sheath failure criterion and metal-water reaction. The mechanical analysis is performed to model the state of stress and strain in the fuel pin subjected to internal and external axial and radial loadings. The strain is normally made up of the following components: elastic, thermal, creep, plastic, fuel swelling and cracks.

The major functions of the Fuel Element Module are: (1) to perform the thermal analyses of fuel pins in a bundle; (2) to simulate the mechanical behaviour of fuel/sheath and fuel bundle; and (3) to calculate the amount of gaseous fission product in the core.
Fuel Channel Module

For the Fuel Channel Module, the following parameters are considered: criterion for pressure and calandria tubes contact, contact dynamics, bundle slumping criterion, metal-water reaction and flow re-distribution effect after bundle slumping.

The major functions of the Fuel Channel Module are: (1) to perform the thermal analysis of the pressure tube and calandria tube; (2) to assess the overall extent of pressure tube ballooning in the core; (3) to calculate the total heat load from the core to moderator; (4) to examine the integrity of the pressure boundary (channel integrity) for all postulated LOCA; and (5) to examine the availability of moderator subcooling.

To use the characteristics of CANDU circuits, each individual reactor channel in each region can be simulated at the same time by using the same header conditions for the fuel channel codes. These boundary conditions at the reactor headers for each core pass are obtained from the reactor system code simulations. Therefore, the overall heat transfer rate between the fuel pins/pressure tubes and the coolant and the header conditions are the most important parameters in the thermalhydraulic module.

Approach at AECL

For the large LOCA calculation, the system response is first obtained with CATHENA using a simplified representation of the core (this could be with between 1 and 10 “average” channels) with less detailed Fuel Element and Fuel Channel Modules. This provides thermalhydraulic conditions at the headers, and a power history for the fuel.

With this information, a detailed thermalhydraulic, fuel element, and fuel channel calculation can be performed for each channel. This is done with a direct linkage between CATHENA and ELOCA (Reference 13) for each fuel element modeled. During the transient run, the information of thermal-hydraulic conditions (including the heat transfer coefficient) for each fuel pin are passed over to the ELOCA code to perform the fuel pin temperature calculations. Then the calculated fuel pin temperature profile is fed back to CATHENA/GENHTP to perform the pressure/calandria tubes temperature calculations. This direct linkage removes the possible errors induced by indirect interface, and models the feedback between all the modules.

Approach at OHN

The approach at OHN is similar to that of AECL for obtaining the system response where TUF is used as the Thermalhydraulic Module. The detailed fuel element and channel behaviour for each channel is then obtained with the HOTSPOT, SMARTT, and FACTAR codes (see Appendix A).
7. BEST ESTIMATE METHODS, VALIDATION, AND UNCERTAINTY

Best Estimate safety analysis relies on the use of computer codes, physical models, correlations and engineering judgment. This approach requires validated methods, which we define as the tools and techniques having specified statements of applicability and accuracy for the specific application, for the selected plant, and for the transient under examination. Validation exercises therefore produce quantified statements of the ranges and associated uncertainties for a specific application and qualify the method for the intended use. Therefore, the statement of accuracy for a particular method is inherently and intimately linked to the chosen safety analysis application.

Since the validation is for specified transients and operating states, the validation must be performed for relevant conditions and designs. This leads directly to the concept of phenomena based validation ‘matrices’ where the method is tested against the ability to predict experimental data, known analytical solutions, or other (numerical and physical) benchmarks.

The validation of Best Estimate analysis codes used in CANADA is performed according to a formal approach described in Reference 14. The uncertainty analysis component is given in a companion paper in this conference (Reference 15).

8. EXAMPLE APPLICATION

CATHENA / ELOCA

An example application, a CATHENA / ELOCA simulation of a postulated LOCA involving a 20% RIH (reactor inlet header) break is given in Reference 13. During the initial stages of a LOCA, the combination of the power pulse due to coolant voiding and the rapid deterioration of the sheath-to-coolant heat transfer rate caused by the loss of coolant, results in sharp increases in the fuel and sheath temperatures. At sufficiently high temperatures, sheath deformation can occur, initially due to the hard contact between the thermally expanding fuel and sheath, and subsequently due to a positive pressure difference across the sheath as the coolant depressurizes. The extent of sheath deformation due to the pressure difference will depend on the initial internal element gas pressure, which is a function of the pre-transient irradiation history, and the coolant depressurization rate.

Two simulations were performed, one with CATHENA alone and the other with CATHENA/ELOCA. The CATHENA-alone simulation was done using a constant fuel-to-sheath heat transfer coefficient of 10 kWm⁻²K⁻¹ and a uniform volumetric heat generation rate across the fuel radius (this has been a standard practice in some CANDU analysis). Results of the calculation are shown in Figures 5 and 6. Note that the response from the BE CATHENA/ELOCA combination is significantly different than the CATHENA-alone calculation.
9. CONCLUDING REMARKS

This paper describes Best Estimate methods developed in Canada for the safety and licensing analysis of the CANDU reactor. AECL and OHN codes have been developed for this application. An important aspect of BE calculations is the coupling of the thermalhydraulic calculations to other disciplines.

An integral requirement for a BE code is a formal validation program as well as the associated uncertainty specification.

REFERENCES

1. CSNI status summary on utilization of best-estimate methodology in safety analysis and licensing, OCDE/GD(97)7, NEA/CSNI/R(96)19, October, 1996.


Figure 2: Typical CANDU Heat Transport System
Figure 4: CATHENA RFSP Coupling Methodology
Figure 5: CATHENA / ELOCA Calculation of Fuel-to-Sheath HTC

Figure 6: Comparison of Fuel Centre-Line Temperatures
APPENDIX: Main Computer codes used in CANDU safety analysis

Many computer codes at OHN and AECL have been utilized in various areas of CANDU safety analysis. For example, the Reactor Fuelling Simulation Programs at AECL (RFSP) and Ontario Hydro Nuclear (OHRFSP) are used in the static 3D calculations of neutron flux, power distribution and neutron balance in a CANDU reactor core. The codes PATRIC (at OHN) and PRESCON2 (at AECL) are used for containment response, FISSCON-II (at OHN) and SMART (at AECL) are used for radiological consequence calculation for a large LOCA safety analysis. In this appendix, only the dynamic codes used in the reactor physics, thermal-hydraulics and fuel channel modules and their functions are mentioned. Detailed physical models and associated numerical methods implemented in each code can be found either in the code manuals or in the papers presented in the literature.

Reactor System Codes

The TUF code (Reference 9), developed at OHN, is a two-fluid code which is currently being used as an analytical tool for operational support and licensing safety analyses for all CANDU reactors (total 12 units) at OHN. It is an upward reactor system code for SOPHT (Reference 16). TUF is a self-contained reactor system code.

The CATHENA code (Reference 3), developed at AECL, is a two-fluid code which is currently being used as an analytical tool for licensing safety analysis for CANDU-6 and proposed CANDU-9 reactors at AECL. Similar to other reactor system codes, CATHENA requires an interface with a package of plant controllers to simulate the LOCA cases.

Reactor Physics Codes

The CERBERUS code (Reference 17), developed at AECL, is based on a quasi-static method to solve the 3D time dependent two-group neutron diffusion equations. The neutron flux is written in terms of a space-time dependent shape function and a space independent amplitude function. The resulting equations are similar in form to the time-dependent diffusion equations but which, in addition to the delayed source terms, contain two sets of time derivatives: the time derivatives of the shape and the time derivatives of the amplitude. The equations are coupled with the amplitude equations which are similar in form to the point kinetics equations. CERBERUS requires a long computing time for a typical large LOCA case. This code has been verified against experiments.

RFSP (Reactor Fuelling Simulation Program, Reference 11) calculates the 3-dimensional flux and power distributions in the reactor core by either of two methods:

- the solution of the finite-difference neutron diffusion equation in two energy groups, or
- a flux-synthesis method using an expansion of the flux in a series of pre-calculated flux modes, with the mode amplitudes calculated to produce the best fit to in-core vanadium-detector readings.

RFSP performs fuel-management calculations and simulates a reactor operating history at specified intervals, taking burnup steps and channel refuellings into account. The code also
performs neutron-diffusion calculations using the "grid-based" local-parameter methodology or the history-based local parameter methodology, in which fuel-bundle properties are dependent on local values of fuel temperature, flux level, coolant density, etc. Using the diffusion-theory method, the program can calculate time-average flux, power and irradiation distributions and channel-refuelling frequencies. RFSP incorporates within it the cell code computational module, which provides the lattice properties for diffusion calculations. The code calculates delayed neutron fractions for 6 precursor groups for each fuel bundle in the core. Additional features of RFSP include bulk-control and spatial-control capabilities; it can calculate (in a quasi-steady state mode) the zone-controller water fills which produce reactor criticality and reproduce a target flux shape in the core.

The SMOKIN code (Reference 18), developed at OHN, is based on one energy group modal expansion technique coupled with a local flux effects correlation technique (Reference 19). The 3D space-time distribution of neutron flux and delayed neutron precursors are expanded as a weighted series of fixed reactor harmonic modes. Applying the Galerkin weighted integration over the reactor core, a set of coupled differential equations are obtained from the mode amplitude weights. Local flux changes induced by the movement of reactivity devices are taken into account through the use of local flux effect functions. The reactor modes are pre-generated by the two-group neutron diffusion code MONIC assuming a nominal core configuration. The nominal configuration is based on the homogeneous representation of the core with reactivity devices at their reference position. SMOKIN is used to cover the wide range of break sizes for LOCA analysis and trip parameter assessment at OHN in the short term (up to five seconds). SMOKIN has been validated against CERBERUS for LOCA type transient and against experiments for operational transient in power reactors.

Fuel Channel/Element Codes

The ELESIM-II/MOD10 code (Reference 20) was developed at AECL and used at AECL and OHN. It is used to provide detailed axial-symmetric thermal/microstructural behaviour for a single fuel element during normal operation conditions. It provides a detailed, mechanistic assessment of the amount and location of fission gas within a fuel element as a function of the power/burnup history. The details of fission gas distribution are then used to assess the gas pressure inside the fuel element.

The ELOCA.Mk4 code (Reference 21) was developed at AECL and used at AECL and OHN. It is used to provide detailed axial-symmetric thermal/mechanical behaviour during a transient for a single fuel element. It provides a mechanistic assessment of the high temperature deformation of the sheath and the effect of oxygen uptake by the sheath on this deformation behaviour. The effect of sheath deformation and pellet thermal expansion/contraction on fuel-to-sheath heat transfer is also modeled.

The HOTSPOT and SMARTT codes (Reference 22) were developed at OHN to provide a detailed (2D, radial and circumferential directions) transient, thermal response for the fuel elements in a symmetric segment of a bundle cross-section. The heat transfer mechanisms modeled are convection, conduction and radiation. It also models the thermal effects of the oxidation of the
zircalloy sheath during a transient. HOTSPOT also provides a one-dimensional assessment of the thermal response of the pressure tube and calandria tube under elevated temperature conditions. The capability to simulate the mechanical behaviour of pressure and calandria tubes contact based on the strain model developed at AECL has been made in the SMARTT code which utilizes the HOTSPOT channel thermal model. Currently the SMARTT code is being used in the safety analysis at OHN.

The CHAN-II/MOD7 code was developed at OHN based on the CHAN-II code at AECL (Reference 23). It is used to provide the transient thermal response of an entire fuel channel during a high temperature condition. The CHAN code series was developed to assess fuel channel behaviour during a postulated accident where cooling is severely degraded. It models the depletion of steam and consequent buildup of hydrogen along a channel due to the zircalloy and steam reaction at elevated sheath temperature. The code models the thermal effects of both pressure tube sagging and ballooning, and bundle slumping onto the pressure tube. The effects of flow redistribution in the channel as a result of these interactions are also accounted for.

The FACTAR code (Reference 24) was recently developed at OHN to unify and to upgrade the fuel channel and element codes (ELESIM and ELOCA code series). It is used to simulate the transient thermal and mechanical behaviour of 37-element or 28-element fuel bundles within a single CANDU fuel channel for any accident conditions that do not result in significant bundle deformation (i.e. extensive bundle slumping). The code has been verified against all fuel channel/bundle results simulated by HOTSPOT, CHAN and TUF codes. A detailed description on the code validation can be found in Reference 25.

The CATHENA code (Reference 3) includes a Generalized Heat Transfer Package (GENHTP), that is able to model heat transfer processes within a CANDU channel in detail. These processes are implicitly coupled to the two-fluid thermalhydraulics. Conduction in the radial and circumferential directions can be calculated for individual fuel elements within a bundle, the pressure tube, and the calandria tube. The effects of thermal radiation, pressure-tube deformation, zirconium-steam reactor, steam starvation, and solid-surface contact (e.g., pressure tube to calandria tube) can all be modeled with the code. Thermal-hydraulic and channel heat transfer behaviour may be tightly coupled with fuel behaviour by using the CATHENA/ELOCA (Reference 13) code suite.
UTILITY PERSPECTIVE ON THE USE OF BEST-ESTIMATE CODES IN THE LICENSING ENVIRONMENT

PREPARED BY P.GARCIA SEDANO

ANKARA JUNE 1998
MODEL DESCRIPTION

SYSTEM MODEL

- RETRAN03 Wide Core Model Based on RETRAN02 Model: extensively validated with plant transients.

- DESCRIPTION:
  - 38 Volumes, 59 Junctions, 12 Core Conductors
  - Plant Control Systems modelated
  - 1D Kinetics Option: 27 separate neutronic regions, Multiple control state model (2 states)
  - Algebraic Slip (Zolotar-Lellouche) and Subcooled Void Model. Consistent with SIMULATE3
  - Model used to support plant operation, revise EPG and issue resolution (ATWS analyses).
IBERDROLA/UITESA APPROACH FOR BWR

* TRANSIENTS ANALYSES METHODOLOGY:

  • BASED ON RETRAN-03

  • INTERMEDIATE APPROACH FOLLOWED, SIMILAR TO OTHER LICENSED METHODOLOGIES.

  • LICENSING PROBLEM: NEED OF AN ACCEPTABLE LICENSING REFERENCE FROM THE REGULATORY BODY IN ORDER TO DETERMINE THE E.M. (TYPICALLY VENDOR LICENSING RESULTS USED).
LOCA CSAU (NUREG/CR-5249)

- All to $2\sigma$
- OpB + adder
- OpB (HTC, HG)
- $M + \sqrt{\sum \hat{\epsilon}}$
- 95% MONTECARLO
- Nominal

Blowdown PCT
First reflood PCT
Second reflood PCT
IBERDROLA/UITESA APPROACH

* BEST ESTIMATE CODES OBTAINED AND PLANT MODELS - DEVELOPED THROUGH PARTICIPATION IN INTERNATIONAL PROJECTS (RETRAN-03/3D FROM EPRI-RETRAN PROJECT, TRAC-BF1/PF1 FROM USNRC-ICAP PROJECT)

* BEST ESTIMATE CODES AND MODELS USED SINCE 1986 FOR OPERATIONAL TRANSIENT ANALYSIS SUPPORT AND PLANT ISSUE RESOLUTION.

* BEST ESTIMATE CODES AND MODELS TO BE USED IN LICENSING APPLICATIONS (RELOAD LICENSING SUBMITTAL, POWER UPRATE LICENSING ....)
* EVALUATION OF LICENSING APPROACHES

- B.E. CODE AND MODEL
  - IDENTIFICATION OF RELEVANT PARAMETERS PHENOMENA FOR THE SAFETY CRITERIA
  - SENSITIVITY ANALYSES/COMPARISON WITH REAL DATA OF RELEVANT PARAMETERS PHENOMENA DETERMINATION OF BIAS + UNCERTAINTIES
    - STATISTICAL TREATMENT OF UNCERTAINTIES
      - Nominal + Bias + 95/95 adder Statistical approach
    - DEFINITION OF E.M. COVERING AT LEAST 95/95 USING SOME VARIABLES AT 2σ AND/OR ADDER
    - DEFINITION OF E.M. USING ALL VARIABLES OF 2σ CONSERVATIVE MODELS
      - Very conservative value Deterministic approach
INTERMEDIATE APPROACH:

- ADVANTAGES: KNOWLEDGE OF RELEVANT PARAMETERS. EASIER TO LICENSE. CAN PROGRESS TO A FULL STATISTICAL APPROACH.

- DISADVANTAGES: SOMETIMES SMALL MARGIN GAINED DUE TO LARGE MODEL BIASES. ADDITIONAL CONSERVATISM TO COVER NOT CONSIDERED VARIABLES.
* EVALUATION OF LICENSING APPROACHES

- DETERMINISTIC APPROACH.
  - ADVANTAGES: EXISTING LICENSING REFERENCES. EASIER TO LICENSE
  - DISADVANTAGES: VERY CONSERVATIVE RESULTS. SMALL OPERATIONAL MARGINS

- STATISTICAL APPROACH
  - ADVANTAGES: CONSERVATISM REDUCED AT A "KNOWN" LEVEL. BETTER KNOWLEDGE OF RELEVANT PARAMETERS.
  - DISADVANTAGES: LACK OF EXISTING REFERENCES. LONGER LICENSING PROCESS. UNCERTAINTIES MUST BE LIMITED IN ORDER TO OBTAINED REASONABLE STATISTICAL VALUES.
* POSSIBLE LICENSING METHODOLOGIES

- DETERMINISTIC APPROACH: CONSERVATIVE E.M. USING BOUNDING VALUES IN ALL RELEVANT VARIABLES AT THE SAME TIME TO PROVIDE CONSERVATIVE RESULTS WITH UNDERTERMINED CONSERVATISM MARGIN. (E.G. LOCA APPENDIX K).

- STATISTICAL APPROACH: B.E. NOMINAL MODEL, WITH DETERMINATION OF UNCERTAINTIES ASSOCIATED TO RELEVANT VARIABLES AND DETERMINATION OF THE LICENSING VALUES AT A FIXED MARGIN (95/95%) (E.G. LOCA CSAU).

LOCA METHODOLOGY

* INITIALLY ORIENTED TO BEST ESTIMATE APPROACH (SECY 83-472)

* FINALLY TURNED TO 10CFR50 APP.K RULES:

    PROS=> EASIER LICENSING
    CONS=> CODE MODIFICATION NECESSARY
* LOCA ANALYSES METHODOLOGY:

- BASED ON TRAC-BF1

- RELAXED APPENDIX K APPROACH FOLLOWING STRICT ACCOMPLISHMENT OF REQUIRED FATURES/CONSERVATISM DEMONSTRATED IN ACCEPTABLE FEATURE.

- INTERMEDIATE APPROACH NOT FOLLOWED BECAUSE OF GREAT AND CAUSE UNDETERMINED UNCERTAINTY OBTAINED IN SOME RELEVANT PHENOMENON RESULTING IN SMALL MARGIN GAIN.
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26/06/98
SENSITIVITY ANALYSIS

INDIVIDUAL UNCERTAINTY

- The change in the ratio of transient DCPR over initial CPR (RCPR) was used to measure the individual uncertainty
- Difference between the RCPR for the perturbated case and the obtained in the base case was used to quantify the uncertainty of each parameter (DRCPR)

MODEL UNCERTAINTY

- Overall uncertainty was obtained combining results of individual uncertainty studies by squares method
- Assumption: Combined uncertainties are normally distributed around their mean
- Upper limit component and equipment were considered in the sensitivity study: is possible to assume that the uncertainty is an upper bound of 95/95 probability level (DRCPR95)
- The change in DCPR with a 95% probability level was calculated
  \[ \text{DCPR95} = [\text{DRCPR95} + \text{RCPR (base)}] \times \text{ICPR} \]
SENSITIVITY ANALYSIS performed to quantify the Aleatory Uncertainty:
- Due to possible deviation between real plant parameters and data included in the model

ORIGIN
- Plant parameters: complex geometry, manufacturing tolerance,....
- Codes: users options, coefficients,...
- Model: variation of relevant plant parameters over their expected values.
- Input: scram speed, gap conductivity, initial conditions,...

PARAMETER CATEGORIZATION
- Parameters with a significant influence on the CPR calculation were included
- Parameters with no know best estimate value or no accepted uncertainty were included with their conservative value
- Significance of parameters was obtained from previous sensitivity analyses or results from other organizations (YAEc, TVA, GPU).

BASE CASE
- Most limiting transient and initial conditions from best estimate analysis: FWCF at ICF-FFWTR
BEST ESTIMATE RELOAD TRANSIENT ANALYSIS (cont.)

INITIAL CONDITIONS: Different points in the expanded power/flow operating map.
  Maximum Extended Load Line Limit (MELL): 102% power and 80% core flow.
  Nominal: 102% power and 100% core flow.
  Increased Core Flow (ICF): 102% power and 108% core flow.
  ICF with Final FeedWater Temperature Reduction (ICFFFFWTR): 102% power, 108% core flow and 250 F degrees as FW temperature.

HOT CHANNEL MODEL
Hot Channel calculations for the different fuel types in Cofrentes Cycle #9
  GE7 - GE10 - GE11 - GE12 - SVEA96

RESULTS
FWCF AT ICFFFFWTR: the most limiting transient
BEST ESTIMATE RELOAD TRANSIENT ANALYSIS

FWCF and GLRWOBP analyzed with the Best Estimate Model

OBJECTIVE: Determination of the transient and initial conditions more limiting

SYSTEM MODEL CHARACTERISTICS:

- Significant parameters on DCPR: implemented with their best estimate value
- NonSignificant or not well define value: Conservative or Technical Specification values used
- Planar gap conductivity
- Realistic scram speed from plant measurements
- Rated initial conditions
## Model Validation

<table>
<thead>
<tr>
<th>TRANSIENT</th>
<th>Parameter</th>
<th>Measurement Error</th>
<th>Maximum Change</th>
<th>Error RETRAN-Plant</th>
</tr>
</thead>
<tbody>
<tr>
<td>Level Setpoint</td>
<td>Level (mm)</td>
<td>60.</td>
<td>107.</td>
<td>91.</td>
</tr>
<tr>
<td></td>
<td>Power (%)</td>
<td>2.</td>
<td>2.8</td>
<td>1.3</td>
</tr>
<tr>
<td>FW Pump Trip</td>
<td>Level (mm)</td>
<td>60.</td>
<td>249.</td>
<td>239.</td>
</tr>
<tr>
<td></td>
<td>Power (%)</td>
<td>2.</td>
<td>26.</td>
<td>28.</td>
</tr>
<tr>
<td>RPLST</td>
<td>Power (%)</td>
<td>2.</td>
<td>55.</td>
<td>53.5</td>
</tr>
<tr>
<td></td>
<td>Core Flow (%)</td>
<td>5.</td>
<td>58.</td>
<td>61.</td>
</tr>
<tr>
<td>TT</td>
<td>Dome pres. (psia)</td>
<td>15.</td>
<td>39.</td>
<td>47.</td>
</tr>
<tr>
<td></td>
<td>Core Flow (%)</td>
<td>5.</td>
<td>58.4</td>
<td>62.4</td>
</tr>
<tr>
<td>GLR</td>
<td>Dome Pres. (psia)</td>
<td>15.</td>
<td>98.</td>
<td>114.</td>
</tr>
<tr>
<td></td>
<td>Core Flow (%)</td>
<td>5.</td>
<td>68.</td>
<td>69.</td>
</tr>
</tbody>
</table>
MODEL VALIDATION (cont.)

* ACCEPTANCE CRITERIA
  
  - Difference between maximum change in RETRAN calculation and plant data must be within measurement uncertainty. If not, the code should be conservative.
  
  - Any safety system activated in the plant transient should be activated in the simulation with the same sequence.

CONCLUSIONS

* ACCOMPLISHMENT OF ACCEPTANCE CRITERIA: Not necessary to consider any systematic uncertainty due to model or code.
MODEL VALIDATION

* SYSTEMATIC UNCERTAINTY associated to the code and model to reproduce plant transients

* PLANT TRANSIENT were selected considering the phenomenology involved in the licensing transients: FWCF and GLRWOBP

* Level Setpoint Change: level decrease due to FW diminution
  - FeedWater Pump Trip: level and power decrease due to FW and inlet subcooling diminution
  - Recirculation Pump Low Speed Transfer: core flow evolution after LPT
  - Turbine Trip: pressure increase and core flow decrease after RPLST
  - Generator Load Rejection with Partial Bypass Failure: pressure increase, SRV behavior, and core flow evolution were checked
MODEL VERIFICATION

Verification of 1D Kinetics System Model with plant transients
* Seven preoperational plant transient included in RETRAN02 verification
* Real plant transients
  – HPCS Injection
  – Core Wide Oscillation
  – Failure in FWCS with loss of FW flow
  – FCV position at startup
* Comparisons with vendor analyses: ATWS calculations

CONCLUSIONS
* The model responds accurately when compared with test data and reference calculations
  * Applicability for simulation of mild and severe transients
PROCESS FOR DEMONSTRATING CONSERVATISM

a) SLICING THE LOCA TIME IN DIFFERENT PERIODS
b) IDENTIFICATION OF THE RANGE OF VARIATION FOR THE MAIN VARIABLES IN EACH PERIOD
c) SELECTION OF THE SUITABLE EXPERIMENTS TO COVER EACH PERIOD
d) ESTABLISHMENT OF THE ACCEPTANCE CRITERIA FOR THE RESULTS CONSERVATISM
e) ASSESSMENT OF THE TRAC ORIGINAL HEAT TRANSFER CORRELATION VS. EXPERIMENTAL DATA
f) IF CONSERVATIVE ➔ ACCEPTABLE
   IF NOT ➔ TRIAL OF OTHER CORRELATION
ACCEPTABLE FEATURES. CONSERVATISM DEMONSTR.

* OTHER CORRELATIONS, NO REQUIRED, ARE ACCEPTABLE ONLY IF CONSERVATISM IS DEMONSTRATED IN THE RANGE OF APPLICATION.

* APP. K QUOTES THE ACCEPTABLE HEAT TRANSFER CORRELATIONS FOR THE DIFFERENT PERIODS OF THE LOCA.

* ALSO IT POINTS THAT ALTERNATE H.T. CORRELATIONS COULD BE ACCEPTABLE AS LONG AS THE COMPARISON BETWEEN EXPERIMENTAL AND CALCULATED H.T. COEFFICIENTS DEMONSTRATES THE CORRELATION CONSERVATISM.
LOCA TIME SLICING

KEY PERIODS OR EVENTS IN A LOCA:

* CRITICAL HEAT FLUX

* POST-CRITICAL HEAT FLUX
  - FILM BOILING
  - STEAM CONVECTION

* SPRAY COOLING
  - FILM BOILING
  - CLAD REWETING
FIGURE 1. COMPARISON OF MEASURED AND CALCULATED CRITICAL QUALITIES
FIGURE 3. COMPARISON OF MEASURED AND CALCULATED HEAT TRANSFER FOR STEAM CONVECTION
FIGURE 4. COMPARISON OF MEASURED AND CALCULATED QUENCHING TIMES
CONCLUSIONS

* A TRAC NEW VERSION HAS BEEN GENERATED: TRAC-BF1/APK.

* Optionally, it can be used in a BEST-ESTIMATE way or in a CONSERVATIVE way.

* The new version complies with both the required and the acceptable features pointed in the 10CFR50 APP.K.

* 21 experiments have been reproduced to demonstrate the conservatism.
* IBERDROLA/UITESA METHODOLOGIES PERSPECTIVES

* FOR BWR:

  - POTENTIAL EVOLUTION OF LOCA METHODOLOGY TO FULL
    CSAU BASED ON MODEL IMPROVEMENTS AND LIMITATION
    OF UNCERTAINTIES.

  - POTENTIAL STATISTICAL APPROACH TO BE USED IN-
    TRANSIENT METHODOLOGY.
* FOR PWR:

- LOCA METHODOLOGY TO BE DEVELOPED BASED ON A PARTIAL CSAU (INTERMEDIATE APPROACH), AS A FIRST STEP.

- TRANSIENT METHODOLOGY: TRACTEBEL METHODOLOGY TO BE USED (AGREEMENT IBERDROLA-TRACTEBEL), BASED ON INTERMEDIATE APPROACH.
**EFFECT OF LARGE UNCERTAINTY**

- **DETERMINISTIC APPROACH:**  \( E.M. \approx N.O.M + \sum \Delta_i \)

- **STATISTICAL APPROACH:**  \( E.M. \approx N.O.M + \sqrt[2]{\sum \Delta_i^2} \)

IS DETERMINED USING THE SAME METHODS

- **IF ONE LARGE UNCERTAINTY (\( \Delta_j \))**
  \[ \sum \Delta_i \approx \Delta_j \]
  \[ \sqrt[2]{\sum \Delta_i^2} \approx \Delta_j \]

\( \Rightarrow \) **DETERMINISTIC EM = STATISTICAL EM**
DEMONSTRATION OF APP. K COMPLIANCE

- IN SOME CIRCUMSTANCES APP. K REQUIRES SPECIFIC MODELS FOR LICENSING

- IN OTHERS, THE CORRELATIONS TO BE USED ARE ACCEPTABLE IF CONSERVATISM IS DEMONSTRATED IN THE WHOLE RANGE OF APPLICATION
REQUIRED FEATURES ADDED

SPECIFIC MODELS FOR:

- DECAY HEAT
- METAL-WATER REACTION
- CRITICAL FLOW MODEL
- NO RETURN TO TRANSITION BOILING AND NUCLEATE BOILING
THE CURRENT STATUS OF THE SAFETY AND THE BEST ESTIMATE ANALYSIS FOR BWR STABILITY, TRANSIENT AND ACCIDENT EVENTS USING THERMAL - HYDRAULICS CODES.

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Abstract
In this paper, the summary of the current situation for the analysis of the stability and abnormal transient/accident events of BWRs using the neutron kinetics and thermal hydraulics codes is presented. In the first part of the paper, the current situation for the stability analysis is presented. In the second part of the paper, the current situation for the analysis of the abnormal transient event is presented. Finally in the last part of the paper, the current situation for the analysis of the reactivity initiated accident is presented.

1. INTRODUCTION

In this paper, the summary of the current situation for the analysis of the stability and abnormal transient/accident events of BWRs using the neutron kinetics and thermal hydraulics codes is presented. The codes are classified into the licensing codes and the best estimate codes. The most of the licensing codes have been originally developed by General Electric Company in United States. However some of these codes have been updated based upon the technical knowledge gained in the research and development, and lots of the operating experience of BWRs in Japan. The best estimate codes are used to support the licensing safety analysis and to have further understanding of the thermal hydraulic phenomena during the stability and abnormal transient/accident events of BWRs.

In the first part of the paper, the current situation for the stability analysis is presented. The stability analysis for the licensing purpose uses K2 code equipped with one point neutron kinetics on frequency domain has been used. In order to qualify K2, the best estimate calculation using three dimensional analysis code TOSDYN in the real time domain were conducted against stability events which were observed at actual plants to compare with the analysis results of by K2. It is concluded that both of K2 and TOSDYN have enough capability to predict stability performance of operating BWRs in Japan.

In the second part of the paper, the current situation for the analysis of the abnormal transient event is presented. The analysis of the abnormal transient event for licensing purpose uses REDY code equipped with one point neutron kinetics and thermal hydraulics.
on real time domain has been traditionally used in Japan. On the other hand, we are aiming to use the three dimensional analysis code TRACG for licensing purpose in near future. In order to qualify TRACG, the reproduction analysis for the transient events conducted at start-up test of Kashiwazaki-Kariwa Nuclear Power Station unit-6 which is the first Advanced BWR in the world were conducted. It is concluded that TRACG has enough capability to predict the overall behavior of important parameters like reactor power, pressure, water level etc., during the abnormal transient event.

In the last part of the paper, the current situation for the analysis of the reactivity initiated accident (RIA) is presented. The analysis of the reactivity initiated accident for licensing purpose uses APEX equipped with two dimensional neutron kinetics have been used. In this part, some result analyzed by ARIES equipped with three dimensional neutron kinetics and thermal hydraulics is also presented.

2. STABILITY ANALYSIS

2.1 Stability Analysis

Three kinds of stability (core, channel and regional stability) are treated in the licensing analysis for BWRs in Japan. The core and channel stability have been traditionally analyzed for the licensing purpose, since BWR was constructed for the first time in Japan. On the other hand, the regional stability was included in the established permitment document very recently. The core stability is core wide neutron flux oscillation which was, for example, actually observed in the stability tests at Peach-Bottom and Vermont-Yankee. The regional stability is out-of phase neutron flux oscillation which was actually observed in the stability tests at Caosoro, KRB-C (Gundremmingen-C), Leibstadt and Ringhals. From stability physics point of view, the core stability is the basic mode and the regional stability is the first mode neutron flux oscillation caused by reactivity feedback effect through a change of void fraction in the core as shown in Figure 2.1.

2.2 Outlines of Stability Analysis Codes

The K2 code is used to analyze stability performance for BWRs, which is based upon the frequency domain with the Laplace transformed functions of linearized equations employing one-point reactor neutron kinetics model and axially one dimensional and multi-parallel channel thermal hydraulic model. The previous K2 code has only a capability to analyze the decay ratio and resonance frequency of the core and channel stability for BWRs. However, it is recently modified so as to be capable to analyze them of the regional stability. In order to analyze the regional stability, a sub-criticality of the core is incorporated into the transfer function of one-point neutron kinetics model of the previous K2 code. The modified K2 is currently used for the stability analysis for the licensing purpose.

\[
G(s) = \frac{1}{s\left(\Lambda + \sum \frac{\beta_i}{s + \lambda_i}\right) + \Delta \rho_i}
\]

\[
\Delta \rho_i = \frac{1}{k_1} - \frac{1}{k_0}
\]

where the sub-criticality of the core (\(\Delta \rho\)) is expressed as a difference of reciprocal numbers of eigen values (k) of the basic and first mode of diffusion equation.

The TOSDYN is the best estimate code developed by TOSHIBA to analyze the spatial effect on stability in the real time domain. Three dimensional neutron kinetics and a parallel thermal hydraulic models are incorporated to calculate the spatial change of neutron flux and coolant flow distribution in the core. TOSDYN can simulate the spatial location of control rods in the core which is important to analyze the regional stability. TOSDYN consists of four component models, the neutron kinetics model of one bundle mesh size, the channel thermal hydraulics model for maximum twenty parallel channel groups, the heat transfer model from fuel rods to coolant and external core model including recirculation flow system and
major control systems. TOSDYN has a capability to evaluate thermal hydraulic behavior in the core during the regional stability.

2.3 Qualification for the Core Stability Analysis

The analysis was conducted to qualify K2 against the core stability tests conducted at Vermont-Yankee and Peach-Bottom. The tests were conducted at relatively high power and low core flow operating points including natural circulation operating point. Figure 2.2/2.3 shows an example of the measured plant data at Vermont Yankee and Peach-Bottom. The decay ratio and resonance frequency were measured and compared to the analysis results by K2 as shown in Figure 2.4. This figure shows that the analysis results by K2 gives conservative decay ratio for the core stability. Figure 2.5 shows the analysis results by TOSDYN for the core stability observed at LaSalle-2 in 1988. The core wide oscillation was initiated by an inadvertent trip of two out of two recirculation pumps. The oscillation was suppressed by the reactor scram on the high neutron flux around 6 minutes later after the trip of two recirculation pumps. It is confirmed that TOSDYN has a enough capability to simulate the core stability observed at LaSalle-2.

2.4 Qualification for the Regional Stability Analysis

It is also confirmed that the modified K2 has a enough capability to analyze the regional stability which were actually observed at the regional stability tests of Caorso, KRB-C, Leibstadt and Ringhals. Figure 2.6 shows the measured LPRM data at Caorso and Figure 2.7 shows the measured LPRM data at KRB-C. As shown in these figures, the regional stability was excited, since out of phase oscillation of LPRM signals were observed in the core. The calculated decay ratios for the regional stability are around 1.1 to 1.2 and it is concluded that the modified K2 has a enough capability to analyze the regional stability. Figure 2.8 shows the comparison of the decay ratio and resonance frequency between the analysis results by TOSDYN and the actual test data measured at Ringhals stability test.

3. TRANSIENT ANALYSIS

3.1 Licensing Analysis by REDY Code

The transient event of BWRs is abnormal plant behavior caused by mis-operation by operators or a single failure of equipment like control systems, pumps and valves. The occurrence frequency of the abnormal transient events is expected to be more than once over the entire plant life. A generator load rejection, an inadvertent tripping of recirculation or feedwater pump, failure of control systems are typical examples of the abnormal transient events for BWRs.

From the safety analysis point of view, the most important parameters during the transient event are the peak values of reactor power, vessel pressure, water level. And MCPR (Minimum Critical Power Ratio) is also important parameter, since it evaluates thermal margin to boiling transition (BT) during the increase of reactor power or decrease of coolant flow.

The REDY code was originally developed by General Electric Company in United States and modified by BWR utilities and vendors in Japan. REDY has modeling for one point neutron kinetics, thermal hydraulics in the core, recirculation pumps, pressure vessel, mainsteam line, control and reactor protection systems. REDY has been used for the licensing analysis and tuned against plenty of data obtained in start-up tests of existing BWRs. From these results, it is shown that REDY has a enough capability to simulate the overall behavior of important parameters during the abnormal transient event.

SCAT is used to calculate the behavior of thermal hydraulic parameters including MCPR in the highest power bundle during the abnormal transient event using boundary conditions which are calculated by REDY. SCAT employs the single channel thermal hydraulics node model in the axial direction, solving the mass, momentum and energy conservation equations and fuel rods heat transfer conduction equations. MCPR is calculated based upon GEXL correlation.
3.2 Kashiwazaki-Kariwa Nuclear Power Station

3.2.1 Outline of the Plant

The Kashiwazaki-Kariwa Nuclear Power Station Unit 6 and 7 are the world’s first Advanced BWRs (ABWR). The site of Kashiwazaki-Kariwa Nuclear Power Station is located in Nigata Prefecture on Japan’s west coast facing the Sea of Japan. Five BWR5 units with an electrical power of 1,100 MWe each are currently in commercial operation at this site. Unit 6 began commercial operation from November 1996 with an electrical power of 1,356 MWe. Unit 7, which is the same design of unit 6, also began commercial operation eight months later after the start of unit 6.

The start-up test program of unit 6 commenced with the first fuel loading at the end of 1995 and demonstrated safe and stable plant operation for all anticipated operating conditions. A large number of variables recorded for each transient test at various power levels provide important data which may be used to validate the advanced design features and models with plant unique characteristics and performance.

The ABWR incorporates advanced technology such as the Reactor Internal Pump (RIP) installed directly at the bottom of the reactor pressure vessel and the Fine Motion Control Rod Drive (FMCRD) achieving a fine motion capability by motor driven control. The RIPs are installed at the bottom of the reactor pressure vessel to simplify the recirculation flow system by eliminating jet pumps and external recirculation loops. The Adjustable Speed Drive (ASD) and the digital control system connected to each RIP controls the recirculation flow rate through the reactor core by controlling the RIP speed. This Recirculation Flow Control System (RFCS) provides quick and stable response of core flow and load following. The FMCRD consists of the electro-hydraulic fine motion control rod drive mechanisms and the Hydraulic Control Unit (HCU) assemblies. The FMCRDs provide electric-motor driven positioning for normal insertion and withdrawal of the control rods and hydraulic-powered rapid control rod insertion (scram) by the HCU for abnormal operating conditions. The FMCRD also provides the Selected Control Rod Run-In (SCRRI) feature as a means of maintaining stability in the reactor. SCRRI is initiated in case of RIP’s trip to prevent the reactor from moving into a operating region of relatively high power and low core flow. The characteristics and performance of these advanced features were extensively analyzed during the design stage and were validated during the start-up test program.

3.2.2 Outline of the Start-up Test Program

In each power condition, various kinds of tests were conducted sequentially beginning with smaller transients such as the control system tuning tests. Large transients like reactor scram event were scheduled during the final part of each condition, assuming that all plant control systems are confirmed to be safe and stable and normal operation has been achieved prior to their initiation. All of the test procedures including cases of malfunction and the test prediction analysis were reviewed and discussed prior to implementation in order to perform the test smoothly and maintain the tight schedule of the start-up test program.

Each start-up test program was selected to confirm safe and stable plant operation to validate advanced design features, to adjust the control system and instrumentation and to store necessary plant data for future operation management and plant design. The recorded data during the abnormal transient tests were evaluated based upon the test judgment criteria which have been prepared prior to the start-up test program. They also specify the necessary action to cope with the unacceptable results.

3.3 Outline of TRACG Code

The actual test data obtained at start-up test program are used for the qualification of TRACG, a BWR transient analysis code. TRACG is the best estimate transient analysis code and has been used to predict the overall behavior during the transient event and to confirm test procedures and design bases. The analysis results by TRACG are compared to the recorded test data in order to evaluate the prediction capability of
TRACG. TRACG is equipped with a three dimensional neutron kinetics model and consists of a flexible system component building model with a node-junction method for the thermal hydraulic calculation.

The TRACG input data for the ABWR have been generated based upon the ABWR geometry of pressure vessel / mainstream line and the latest design specifications as shown in Figure 3.1. These data have been applied since 1994 to simulate the overall behavior during the abnormal transient event in the start-up test program. And it has been updated with respect to the actual nuclear characteristics, plant heat balance, control systems and setpoints of transient mitigation function in order to improve prediction accuracy. TRACG includes the modeling for plant control systems, such as the Recirculation Flow Control System (RFCS), the Feedwater Control System (FWCS) and the Steam Bypass and Pressure Control System (SBPCS). The analysis results by TRACG and the actual test data for a three RIPS trip, generator load rejection and SCRRR insertion test are presented in the next section. These transients are typical in the ABWR start-up test program.

3.4 Qualification Analysis for TRACG
3.4.1. Three RIPS Trip Test

No more than three RIPS are connected to any one electrical power supply bus. Thus, the worst single failure can only cause three RIPS to trip. In the start-up test program, one to three RIPS trip tests were conducted. The three RIPS trip test obviously showed the largest transient in the trip tests. Figure 3.2 shows a comparison between the analysis results by TRACG and the actual test data. If a trip of three RIPS is caused by a single failure in an electrical power supply, the core flow shows a rapid reduction in only a few seconds and then reaches a stable flow corresponding to the remaining RIPS operation. The rapid coastdown of core flow generates more voids in the reactor, which reduces the reactor power due to negative feedback of void reactivity and also causes a temporary swelling of water level. It is mitigated by the water level control action of FWCS and establishes a new steady state at the same initial level condition. As shown in Figure 3.2, the analysis results by TRACG shows fairly good agreement with the actual test data.

3.4.2 Generator Load Rejection Test

When electrical grid disturbance occurs and results in loss of electrical load on the turbine-generator, fast closure of the turbine control valves (TCVs) is initiated by the power and load unbalance relay (PLU) which initiates a reactor scram and a trip of four out of ten RIPS simultaneously.

Figure 3.3 shows the comparison between the analysis results by TRACG and the actual test data for the generator load rejection from rated power condition. The fast closure of TCVs causes a rapid reduction of turbine steam flow, and results in an increase of vessel pressure. This increase of vessel pressure is mitigated by fast opening of relief valves and turbine bypass valves (TBVs). A lower stable pressure is achieved with closure of all TBVs controlled by the SBPCS. Within a few minutes of generator load rejection, the mild decrease of vessel pressure occurs due to steam condensation at the exit of steam separators. This is a result of interfacial heat transfer between subcooled liquid flow through the separators and saturated vapor in the vessel dome, and a corresponding reverse steam flow at the exit region of steam separators.

Core flow initially decreases by four RIPS trip and automatic runback of remaining six RIPS and becomes subcooled due to the reactor scram. It is then heated gradually by decay heat and warmer water entering the core region due to the shutdown of cold feedwater flow. Core flow becomes saturated again and vessel pressure increases slowly and settles back to the steady state. The generator load rejection also causes a relatively large water level drop due to the initiation of reactor scram. However, the water level drop allowed enough margin to the reactor isolation setpoint by the water level control response of FWCS. As shown in Figure 3.3, the analysis results by TRACG shows overall good agreement with the actual test data.
3.4.3 SCRRRI Insertion Test

SCRRRI is introduced as a core stability countermeasure. When some RIPS trip occurs and results in operating state conditions at relatively high power and low core flow where the core stability is a concern, predetermined control rods are automatically inserted to avoid an unstable state by reducing reactor power. The manual SCRRRI insertion test was performed at the 50% reactor power condition during the start-up test program. Figure 3.4 shows the comparison between the analysis results by TRACG and the actual test data.

The selected control rods are inserted by the drive motor and it takes approximately two minutes for full insertion. The transient behavior of this event is mild. Reactor power decreases gradually with the SCRRRI insertion. During the reactor power reduction, void fraction in the core region becomes lower and causes water level to move downward. The feedwater flow, regulated by the FWCS, responds in order to maintain constant water level and constant inventory in balance with mainsteam flow. Thus, feedwater flow decreases but remains higher than mainsteam flow. As a result of this response, water level increases slightly after the SCRRRI insertion due to a certain delay of the water level control response. This delay is adjusted in order to maintain stable operating conditions. The analysis results by TRACG in Figure 3.4 shows good agreement with the actual test data as well as showing the ability of TRACG to simulate a SCRRRI transient involving changes in power distribution.

From the transient events of the start-up transient test program at Kashiwazaki-Kariwa Nuclear Power Station unit 6, many valuable test data were obtained. TRACG has been qualified based upon the actual test data. Consequently, it is confirmed that TRACG is fully capable of accurately predicting transient response and will be useful for application to the plant design for existing BWRs and future ABWRs.

Unit 6 and 7 are operating well without any significant problem after the start of their commercial operation as the world’s first ABWR.

3.5 Application to Licensing Analysis of TRACG

The current licensing analysis for the abnormal transient event uses REDY and SCAT equipped with one dimensional neutron kinetics. However, Japanese BWR utilities and vendors are aiming to upgrade the analysis for the abnormal transient event by applying TRACG for the licensing analysis. In the licensing analysis for BWRs in Japan, it is requested that the results by analysis codes shall be conservatively evaluated considering uncertainty of input data and/or expected operating conditions. The analysis by REDY and SCAT for the abnormal transient event assumes conservative inputs data. For example, more negative void reactivity coefficient or more slower scram insertion speed are assumed in order to conservatively evaluate the peak power, vessel pressure or MCPR reduction ($\Delta$MCPR) for the analysis of the generator load rejection.

On the other hand, the analysis by TRACG equipped with three dimensional neutron kinetics uses basically the best estimate values for its input data. Therefore, the calculated results by TRACG gives the best estimate peak values of reactor power, vessel pressure or $\Delta$MCPR during the transient event. In order to conservatively evaluate the results of the analysis by TRACG considering uncertainty of input data, the sensitivity analysis method by using statistical approach is currently under study. In this calculation, input data which might have uncertainty are simultaneously and randomly sampled. And then Goodness-of-fit test of the normal distribution for the result is evaluated. From this result, the combined uncertainty is calculated so as to determine the upper bound of the results with 95% probability and 95% confidence.

One example is shown in Figure 3.5. The sensitivity analysis for $\Delta$MCPR during the generator load rejection event is performed by changing input parameters like void reactivity coefficient, void fraction and gap-conductance, etc. Such a statistical approach is now under study aiming to apply TRACG to the licensing analysis in near future.
4. ANALYSIS FOR REACTIVITY INITIATED ACCIDENT

4.1 The Reactivity Initiated Accident

The reactivity initiated accident (RIA) is defined as the accident in which the reactivity more than one dollar is inserted in the core which causes the prompt critical. It is one of design basis accidents for BWRs in Japan as well as loss of coolant accident (LOCA). The reactivity initiated accident is caused by a postulated drop of control rods (RDA) from the core or an inadvertent rod withdrawal error (RWE).

In BWRs, the reactivity initiated accident causes a rapid increase of reactor power, but it is mitigated by Doppler and moderator density feedback effects. From the safety analysis point of view, it is required to confirm the coolable geometry, the safe shut down and the integrity of pressure boundary.

4.2 The Current Status for RIA Analysis by APEX

APEX is used to calculated this accident for the licensing purpose in Japan. APEX calculates the change of an average power using one point neutron kinetics, and a change of power distribution based upon two dimensional diffusion model. APEX employs a conservative assumption by neglecting moderator density feedback effect. APEX can not treat the spatial distribution of fuel pellet exposure. SCAT calculates the increase of enthalpy of fuel pellets during the reactivity initiated accident using output of APEX.

4.3 The Best Estimate Analysis by ARIES

The Japanese BWR utilities and vendors have jointly developed the best estimate analysis code ARIES to calculates the reactivity initiated accident. ARIES has a three dimensional neutron kinetics coupled with multi-channel thermal hydraulic effect associated with the local power increase due to control rods movement. The basic thermal hydraulic model of ARIES is a non equilibrium model for separated two phase flow which uses the drift flux correlation to calculate void fraction in the core.

Figure 4.1 show the typical analysis results by ARIES for the rod drop accident initiated from cold critical condition. The reactor power increases to around ten to eight times of initial value within one second. However, it is suppressed by Doppler and moderator density feedback effect. The maximum net reactivity inserted in the core is much less than two dollars. And Figure 4.2 shows the typical analysis results by ARIES for the rod drop accident initiated from hot stand-by condition.

If the rod drop accident is initiated, the fuel rods would experience coolant boiling transition (BT) immediately and the fuel pellet enthalpy increases as shown in Figures. According to the analysis results by AREIS, the peak fuel pellet enthalpy is less than the maximum allowable value defined in Japanese safety evaluation guide line. Since the threshold of the peak fuel pellet enthalpy which violates fuel integrity depends upon fuel pellet exposure as shown in Figure 4.3, it is much more important to calculate three dimensional exposure distribution.

ARIES has been verified with the benchmark problems proposed by OECD/NEA/CSNI and the test data conducted at SPERT-3 E core. Figure 4.4 shows comparison between the analysis results by AREIS and DANO against the rod drop accident proposed by OECD/NEA/CSNI. The power history calculated by ARIES has a similar trend calculated by DENO code, since both code use almost the same conditions for solving the problems. Both code solve the one-energy group neutron equation with almost the same mesh spacing. Figure 4.5 shows comparison between the analysis results by ARIES and the actual test data at SPERT-3 E core. The SPERT-3 reactor is small,oxido-fueled, pressurized water reactor that, except, for size, is generally characteristic of a commercial light water reactor with essentially no fission product inventory in the core. The E-core has 60 fuel assemblies. Most of the fuel rods are contained in forty eight 3 × 3 inches square assemblies that contain 25 rods in 5 × 5 rectangular array. From these results, it is confirmed that the capability of ARIES is fairly nice.
5. CONCLUSION

Three kind of codes equipped with one point neutron kinetics are used for the licensing analysis in Japan, K2 for the stability analysis, REDY/SCAT for the abnormal transient analysis and APEX/SCAT for the reactivity initiated accident. On the other hand, three kind of codes equipped with three dimensional neutron kinetics have been developed to perform the best estimate analysis, TOSDYN for the stability analysis, TRACG for the abnormal transient analysis and ARIES for the reactivity initiated accident. Each code calculates the behavior of BWRs in the real time domain. The major difference of each code is shown in Table-2. For each code, the qualification study have been conducted and it is confirmed that each code have fairly good performance to simulate the events in BWRs. Each code is currently used for the best estimate analysis to support the licensing analysis. However, Japanese utilities and vendors are aiming to apply these codes to the stability and transient/accident events to upgrade the licensing analysis in near future.

Acknowledgments

The authors would like to acknowledge the contribution of all those who were involved in the execution for making this paper.

References


### Table-1  Licensing and Best Estimate Analysis Codes

<table>
<thead>
<tr>
<th>Stability Analysis</th>
<th>Licensing Analysis</th>
<th>Best Estimate Analysis</th>
</tr>
</thead>
<tbody>
<tr>
<td></td>
<td>One point Neutron Kinetics</td>
<td>Three Dimensional Neutron Kinetics</td>
</tr>
<tr>
<td>Stability Analysis</td>
<td>K2, HIBLE, FABLE</td>
<td>TOSDYN, STANDY</td>
</tr>
<tr>
<td>Analysis for Abnormal Transient</td>
<td>REDY/SCAT</td>
<td>TRACG</td>
</tr>
<tr>
<td>Analysis for Reactivity Initiated Accident</td>
<td>APEX/SCAT</td>
<td>ARIES</td>
</tr>
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</table>

### Table-2  Comparison of Three Dimensional Codes

<table>
<thead>
<tr>
<th>Neutron Diffusion Equation</th>
<th>TOSDYN</th>
<th>TRACG</th>
<th>ARIES</th>
</tr>
</thead>
<tbody>
<tr>
<td>Neutron Diffusion Equation</td>
<td>Modified one Group Approximation</td>
<td></td>
<td></td>
</tr>
<tr>
<td>Neutron Kinetics Model</td>
<td>Prompt Jump</td>
<td>Improved Quasi-Static Approximation</td>
<td></td>
</tr>
<tr>
<td>Thermo Hydraulics Model</td>
<td>5 Equations + Drift Flux Model</td>
<td>6 Equations + Drift Flux Model</td>
<td></td>
</tr>
<tr>
<td>Number of Channel Types</td>
<td>20</td>
<td>99</td>
<td>120</td>
</tr>
<tr>
<td>Fuel Heat Transfer Model</td>
<td>One Dimensional for Radial Direction</td>
<td></td>
<td></td>
</tr>
<tr>
<td>Number of Fuel Cells</td>
<td>15</td>
<td>-</td>
<td>100</td>
</tr>
<tr>
<td>Model for Ex-core</td>
<td>Detailed Model including Control Systems</td>
<td>Simple Model</td>
<td></td>
</tr>
<tr>
<td>Application</td>
<td>Stability</td>
<td>Abnormal Transient</td>
<td>RIA</td>
</tr>
</tbody>
</table>
The Core Stability
(Basic Mode Oscillation)

The Regional Stability
(First Mode Oscillation)

Figure 2.1 The Core and Regional Stability

Vessel Pressure
Core Pressure
APRM-B
Water Level
Feedwater-A (DP)
Mainsteam Flow
Total Core Flow
Core Plate ΔP

Time (sec)
0 4 8 12 16 20 24 28
6.68 MPa
6.73 MPa
51.4 %
0.86 mH₂O
15.7 kPa
378 kg/s
1972 kg/s
5.52 kPa

Figure 2.2 The Stability Test at Vermont-Yankee
Figure 2.3  The Stability Test at Peach-Bottom

Figure 2.4  The Comparison of Core Decay Ratio and Resonance Frequency for Vermont-Yankee and Peach-Bottom Stability Test
Figure 2.5  The Analysis Results by TOSDYN for the Core Stability Event at LaSalle-2

Figure 2.6  The Stability Test at Caorso
Figure 2.7  The Stability Test at KRB-C

Figure 2.8  The Comparison of Regional Decay Ratio and Resonance Frequency for Ringhals Stability Test
Figure 3.1 TRACG System Component Model for ABWR Analysis

Figure 3.2 Comparison of the Analysis Results by TRACG and the Actual Test Data
(Three RIPs Trip Test)
Figure 3.3  Comparison of the Analysis Results by TRACG and the Actual Test Data
(Generator Load Rejection Test)

Figure 3.4  Comparison of the Analysis Results by TRACG and the Actual Test Data
(SCRRJ Insertion Test)
Figure 3.5 TRACG Sensitivity Study for $\Delta$MCPR Distribution

Figure 4.1 The Analysis Results by ARIES for the Rod Drop Accident from Cold Critical Condition
Figure 4.2  The Analysis Results by ARIES for the Rod Drop Accident from Hot Stand-by Condition

Figure 4.3  Fuel Pellet Exposure and Enthalpy
Figure 4.4  The Comparison of the Analysis Results by ARIES and DANO for the Rod Drop Accident

Figure 4.5  The Comparison of the Analysis Results by ARIES and the Actual Test Data at SPERT-III E-core
SESSION III - PART 2:
BEST-ESTIMATE METHODOLOGIES AND ASSOCIATED UNCERTAINTIES
Application of New BE Code to PWR and APWR LBLOCA Analysis

Presentation at
OECD/CSNI SEMINAR ON
BEST ESTIMATE METHODS IN THERMAL HYDRAULIC SAFETY ANALYSIS
29 June - 1 July

Authors:  S.Urata    The Kansai Electric Power Co., INC
          K.Okabe    Mitsubishi Heavy Industries, LTD.

Contents

1. Background
2. Current Strategy for BE code application
3. Verification plan
4. Summary
1. Background
1-1 Licencing Plan

- APWR will be constructed
  - Features of APWR are attached
- High burnup fuel (up to 55 GWD/t) will be introduced to existing PWRs and APWR

- Introducing high burnup fuel has potentials of higher Peak Clad Temperature (PCT) at LOCA although margin to limit value (1200 °C) still exists
- Rational LOCA evaluation is expected
  - Flexibility of core management
  - Future uncertainties
  - Public acceptance

---

**APWR in Japan (1/2)**

<table>
<thead>
<tr>
<th>Item</th>
<th>APWR</th>
<th>Current 4 Loop (Tsuruga-2)</th>
</tr>
</thead>
<tbody>
<tr>
<td>Electric Output (MWe)</td>
<td>Approx. 1,530</td>
<td>1,160</td>
</tr>
<tr>
<td>NSSS Output (MWt)</td>
<td>Approx. 4,450</td>
<td>3,423</td>
</tr>
<tr>
<td>Core</td>
<td></td>
<td></td>
</tr>
<tr>
<td>Fuel</td>
<td></td>
<td></td>
</tr>
<tr>
<td>Fuel Assembly</td>
<td>17 × 17</td>
<td>17 × 17</td>
</tr>
<tr>
<td>Fuel Length (m)</td>
<td>257</td>
<td>193</td>
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<tr>
<td></td>
<td>3.66</td>
<td>3.66</td>
</tr>
<tr>
<td>Radial Reflector</td>
<td>Stainless Steel</td>
<td></td>
</tr>
<tr>
<td>Number of Loops</td>
<td>4</td>
<td>4</td>
</tr>
<tr>
<td>RV</td>
<td></td>
<td></td>
</tr>
<tr>
<td>Diameter (m)</td>
<td>5.2</td>
<td>4.4</td>
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<td>Height (m)</td>
<td>13.6</td>
<td>12.9</td>
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<tr>
<td>SG</td>
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<tr>
<td>Heat Transfer Area (m²)</td>
<td>6,500</td>
<td>4,780</td>
</tr>
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<td>Tube Material</td>
<td>Inconel TT690</td>
<td>Inconel TT600</td>
</tr>
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<td>Containment (Type)</td>
<td>PCCV</td>
<td></td>
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<tr>
<td>Turbine Generator (Type)</td>
<td>TC6F52</td>
<td>TC6F44</td>
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### APWR in Japan (2/2)

<table>
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<th>Item</th>
<th>APWR</th>
<th>Current 4 Loop</th>
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</thead>
<tbody>
<tr>
<td>MOX Fuel Loading</td>
<td>≥1/3 Core</td>
<td>1/4～1/3 Core</td>
</tr>
<tr>
<td>High Burn up Fuel Loading</td>
<td>≥55GWD/t</td>
<td>48GWD/t</td>
</tr>
<tr>
<td>Occupational Dose</td>
<td>0.2 (Man·Sv/Year)</td>
<td>0.4～5 (Man·Sv/Year)</td>
</tr>
<tr>
<td>Design Lifetime</td>
<td>60 Years</td>
<td>40 Years</td>
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<tr>
<td>Core Damage Frequency</td>
<td>Approx. 1 decade lower</td>
<td>base</td>
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<tr>
<td>Maintenance Outage</td>
<td>&lt; 40 days</td>
<td>&lt; 50 days</td>
</tr>
</tbody>
</table>

### Core and Reactor Vessel

- Rod Control Cluster
- Reactor Vessel
- Baffle Former
- Radial Reflector

257 Fuel Assemblies

APWR

193 Fuel Assemblies

Current 4 Loop
The Advanced Accumulator

**N₂ Gas**  
Water  
Stand Pipe  
Vortex Damper  
Large Flow Injection

**N₂ Gas**  
Water  
Vortex Damper  
Small Flow Injection
1-2 Technical Background for further improvement (1/2)

- 2D/3D Project was finished and supplied fruitful data/base

2D/3D project
- Japan/Germany/US cooperation project
- Major role of each country
  Japan: Large scale experiments
  Cylindrical Core Test Facility (CCTF)
  Slab Core Test Facility (SCTF)
  1/22 volumetric, 1/1 height ratio for 4loop Japan/US PWR

  Germany: Large scale experiments
  Upper Plenum Test Facility (UPTF)
  1/1 volumetric, 1/1 height ratio for 4loop German PWR
  Refill related data are useful for Japan/US PWR

  US: TRAC code calculation
  Advanced two-phase instrumentation

1-2 Technical Background for further improvement (2/2)

- Computer technology development enable us to use multi-dimensional two-phase computer code

- At US, statistical method with BE (WCOBRA/TRAC) code to evaluate 95% confidence level PCT was approved as licensing calculation.

-WCOBRA/TRAC code
  multi-dimensional, non-equilibrium, 3 field (liquid, drop, vapor) for reactor vessel
  one-dimensional, non-equilibrium, drift-flux model for loop portion
1-3 Current LOCA Evaluation Model (EM) of Japanese PWR industries

Large Break Loss of Coolant Accident (LBLOCA) EM was established at 1980s:
- For Blowdown Phase
  One dimensional Model
  Drift Flux Model was improved based on industries experiments (SATAN-M code)
- For Refill Phase
  One dimensional Model
  Conservative Bypass Deficit Model
- For Reflood Phase
  One dimensional Model
  Detailed physical modelling (BASH-M code)
  Verified by JAERI large scale reflooding experiment (CCTF: part of 2D/3D experiment)
- Decay heat: 1979 ANS model with 2σ margin

2. Current Strategy for BE code application
2-1 Basic approach

- BE code and statistical treatment for uncertainties are excellent approach theoretically for safety evaluation.
- However, at current state, Japanese PWR industries still select conservative approach from the practical reason.
  - conservative model from the view of PCT
  - conservative input

For this approach, by applying conservatism to WCOBRA/TRAC code, MCOBRA/TRAC is developed
- model modification to add additional safety margin
- model improvement from the point of BE capability
2-2 Code modification for MCOBRA/TRAC (1/2)

- From experiments calculation for blowdown, refill and reflood phase, modification to reflood phase model is judged to be done to add safety margin.

- WCOBRA/TRAC core heat transfer model at reflood phase is modified to calculate higher peak clad temperature (PCT) than results of large scale system effect tests.

2-2 Code modification for MCOBRA/TRAC (2/2)

- Others
  - For decay heat, correlation of Atomic Energy Society of Japan (AESJ) with margin(uncertainty) is used.
  - Modification for BE capability
    Drift flux model is modified to MHI model used in current EM codes because it is well verified by Japanese industries experiment
3. Verification plan (1/2)

- MCOBRA/TRAC capability is being verified by large scale experiments such as

  For BLOWDOWN phase : LOFT, ROSA  
  REFILL phase : UPTF, CCTF  
  REFLOOD phase : CCTF, SCTF

and by other small scale experiments.

3. Verification plan (2/2)

- Efforts for developing MCOBRA/TRAC are focused on the effect of modification and below results are expected
  - important phenomena for LOCA evaluation such as items in attached table are well evaluated
  - conservative results from the view of PCT

- And also, LOCA sensitivity studies for typical PWR are able to specify the conservatism of input data set
### Experiments for verification

<table>
<thead>
<tr>
<th>Important Phenomena</th>
<th>Experiments</th>
<th>Marviken</th>
<th>LOFT</th>
<th>ROSA II</th>
<th>UPTF</th>
<th>JAERI Large Scale Reflood</th>
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<td></td>
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<td></td>
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<td></td>
<td>SCTF</td>
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<td></td>
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<td>Critical Flow</td>
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<tr>
<td>Refill</td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td>×</td>
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<tr>
<td>Counter Current</td>
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<td></td>
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<tr>
<td>Flow at Downcomer</td>
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<td></td>
<td></td>
<td></td>
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<tr>
<td>Reflood</td>
<td></td>
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<td>×</td>
<td>×</td>
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<td>multi dimensional</td>
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<tr>
<td>Core heat transfer</td>
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<td>×</td>
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<td>Entrainment at upper</td>
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<td>×</td>
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<tr>
<td>Steam binding at SG</td>
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<td></td>
<td></td>
<td>×</td>
<td></td>
<td></td>
</tr>
</tbody>
</table>

*(notes): small scale experiments are deleted in this table*

---

### Verification by large scale reflood test facility (CCTF)

**Birds eye’s view**

**CCTF (Cylindrical Core Test Facility)**
- largest scale in the world
- Compared with current 4 loop PWR
- 1/1 for height, 1/20 for fluid volume

**Run62 (Base Test)**
- Pressure: 0.2 MPa
- Core Power: 9.365 MW
- Average linear power: 1.40 kW/m
4. Summary (1/2)

- Introduction of high burnup fuels to APWR and existing PWRs is planned.

- Japanese PWR industries are developing new LBLOCA evaluation code, MCOBRA/TRAC, which is the deterministic and conservative approach at current stage.

- For this evaluation, BE code (WCOBRA/TRAC) is modified (MCOBRA/TRAC).
  - Core heat transfer coefficient at some region of WCOBRA/TRAC is reduced
  - Inputs for MCOBRA/TRAC are the same conservative values as current analysis.
4. Summary (2/2)

- Verification of MCOBRA/TRAC is being done by reflecting 2D/3D projects (CCTF/SCTF, UPTF) data and other experiments.

- Plant sensitivity studies by MCOBRA/TRAC will specify the conservatism of input data set.
APPLICATION OF BEST-ESTIMATE METHODS FOR
OPTIMIZATION OF EPR'S EMERGENCY
CORE COOLING MODE

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F-92064 PARIS LA DEFENSE CEDEX
CONTENTS

1 Purpose of the study

2 EPR's safety injection system – evolution and features

3 Analysis methodology
   3.1 General description
   3.2 Computer codes used
   3.3 Qualification and verification

4 Analysis results
   4.1 Initial and boundary conditions
   4.2 Double-ended guillotine break
   4.3 Other loss-of-coolant accidents

5 Conclusions
1 PURPOSE OF THE STUDY

It is shown how best-estimate analysis methodologies were applied to the optimization of EPR's emergency core cooling mode, and which were the benefits.

EPR's SIS comprises 4 trains, each of them consisting of one MHSI, one accumulator and one LHSI. Injection mode is into the cold legs of the main coolant line for the short term.

This injection mode is usual in French PWRs. For German PWRs only combined ECC injection (i.e. into both the cold leg and the hot leg of the main coolant line) was used up to now.

During the developmental process of EPR's ECC mode, extensive thermal-hydraulic analyses have been performed for analyzing the impact of each ECC mode on ECC efficiency as well as on containment behavior during representative LOCA.

The present paper describes the evolution from combined injection to cold leg injection and summarizes results of comparative analysis.

Note that studies presented below have been performed at the very beginning of EPR basic design. Therefore, plant data, initial and boundary conditions as well as some assumptions regarding balance of plant used for analyses may differ from those, which will finally apply for EPR.
2 EPR’S SAFETY INJECTION SYSTEM
EVOLUTION AND FEATURES

EPR’s safety injection system (SIS) comprises 4 trains, each of them consisting of:

- one medium head safety injection (MHSI)
- one accumulator
- one low head safety injection (LHSI)

Injection mode is into the cold legs of the main coolant line (so-called “cold-leg injection”).

Compared to SIS of N4 PWR, EPR’s SIS has the following improved features:

- fourfold redundancy
- strict separation of redundancies
  - no headers
  - systematic consideration of preventive maintenance possible
- reduced shut-off head of MHSI-pumps
  - 8.0 MPa instead of 14.5 MPa
2 SAFETY INJECTION SYSTEM
EVOLUTION AND FEATURES (CONTD.)

Compared to SIS of KONVOI, EPR’s SIS has the following improved features

- Reduced shut-off head of MHSI-pumps
  - 8 MPa instead of 11 MPa

- Increased accumulator pressure
  - 4.5 MPa instead of 2.6 MPa

- Optimized accumulator water/nitrogen volume ratio
  - 32/15 instead of 36/11

- Increased shut-off head of LHSI-pumps
  - 2 MPa instead of 1 MPa

- Simplified system design
  - no accumulator isolation
  - In-containment refueling water storage tank (IRWST) instead of water tanks
  - switch over to suction from sump not necessary
  - automatic complete secondary side cooldown replaced by partial cooldown

- Optimized capacity
  - 4 instead of 8 accumulators
  - reduced MHSI-pump capacity
  - reduced LHSI-pump capacity
EPR safety system configuration - comparison against N4 and KONVOI PWR
Nuclear Power International

EPR primary side safety systems

Evolution of EPR ECC-mode

2 SAFETY INJECTION SYSTEM
EVOLUTION AND FEATURES (CONT'D.)

The EPR ECC-mode with cold leg injection is the outcome of an evolutionary process, which mainly had two developmental steps:

1. From KONVOI SIS, i.e. hot leg HPSI, hot and cold leg accumulator injection and hot and cold leg LHSI (so-called combined injection), to an EPR ECC-mode with cold leg MHSI, hot leg accumulator injection and combined LHSI (so-called hot leg injection), and

2. From hot leg injection to cold leg injection

The main change between these steps is the number and location of accumulators.

Since accumulators are principally designed for mitigation of LBLOCA, the evolution from combined injection to cold leg injection was accompanied by LBLOCA calculations.
3 ANALYSIS METHODOLOGY

3.1 GENERAL DESCRIPTION

- Siemens realistic methodology for LOCA analyses is used. It is a best-estimate analysis with appropriate conservative initial and boundary conditions.

- Siemens Realistic Methodology is assessed against appropriate separate and integral effect tests.

- Prediction uncertainties are quantified using the Code Scaling, Applicability and Uncertainty (CSAU) Evaluation Methodology, which provides a logical process for assessing and quantifying uncertainties of a computer code with respect to LBLOCA phenomena.
3.2 COMPUTER CODES USED

- Siemens Realistic LOCA Evaluation Model is based on
  - S-RELAP5 for thermal-hydraulics of RCS
  - COCO for containment thermal-hydraulics
  - S-RELAP5 and COCO coupled via EUMOD-interface

THE COMPUTER CODE S-RELAP5

- The thermal-hydraulic code S-RELAP5 was developed by Siemens Power Corporation (SPC), USA, for performing realistic analysis of LOCA for PWRs.

- It is a RELAP5-based thermal-hydraulic system code which incorporates features of RELAP5/MOD2 and RELAP5/MOD3, as well as improvements made by Siemens.

- In general, the improvements and modifications included are those required to provide congruency with the unmodified literature correlations and those required to obtain adequate simulation of key LBLOCA experiments.

- The code structure for S-RELAP5 was modified to be essentially the same as that for RELAP5/MOD3, with the same code portability features. The coding for reactor kinetics, control system and trip systems was replaced by those of RELAP5/MOD3.
3.2 COMPUTER CODES USED (CONTD.)

THE CONTAINMENT CODE COCO

- The computer code COCO was developed by Siemens for analysis of temperature and pressure evolution in a PWR dry containment under LOCA conditions. The code is also used for containment design (peak pressure calculation).

Main features

- lumped parameter model with balance equations for mass, volume and energy
- containment divided into two subsystems: the atmosphere inside the building and the sump
- thermal non-equilibrium between the subsystems as well as mass and heat exchange modelled
- heat can be exchanged with adjacent walls or containment internals
- balance of incoming and outgoing mass and energy fluxes
3.2 COMPUTER CODES USED (CONT'D.)

THE COUPLING INTERFACE EUMOD (EXTERNAL USER MODELS)

- The interface EUMOD is a set of generalized data transfer subroutines for coupling of RELAP5 with any other FORTRAN program.

Main features

- it transfers RELAP5 variables to an external code
- it transfers variables generated by the external code back to RELAP5
- the interface does not interfere with the RELAP5 algorithm
- it allows redefinition of certain RELAP5 input data after each time step
- for each time step, RELAP5 calls EUMOD which then performs the necessary data transfers between the two codes and calls for execution of the external code
- after execution of the external code, control is returned to RELAP5, which continues execution
3.2 COMPUTER CODES USED (CONTD.)

S-RELAP5/COCO - COUPLING METHODOLOGY

- Transfer of specific thermal-hydraulic variables to the COCO code at any successfully terminated RELAP5 time step
  - break mass flow rates
  - break enthalpy flow rates
  - enthalpy/mass flow rates of other injections into containment
  - size of the time step used last
  - containment pressure used during the last RELAP5 time step

- COCO computes the new containment pressure and temperature based on these data

- Updated boundary conditions are transferred to S-RELAP5 and used for computation of the new RELAP5 time step
3.2 COMPUTER CODES USED (CONTD.)

S-RELAP5/COCO - COUPLING METHODOLOGY

**RELAP5**
- INITIAL CONTAINMENT PRESSURE PCOLD
- RELAP5-INPUT FOR LB-LOCA
- RELAP5 CALCULATION

**MANUAL DATA TRANSFER**
- MASS AND ENTHALPY FLOW THROUGH THE BREAK
  - PCOLD = PCNEW

**COCO**
- COCO INPUT
- COCO CONTAINMENT PRESSURE CALCULATION
- CONTAINMENT PRESSURE PCNEW

**FLOW DIAGRAM**
- NO: PCNEW > PCOLD
- YES: CONTINUE

STOP

COCO-Schema/Cu,5mv28.10.92

---


379
3.3 QUALIFICATION AND VERIFICATION

S-RELAP5

Capabilities of S-RELAP5 were intensively assessed against appropriate IET (e.g. CCTF run 54, CCTF run 79, LOFT L2-6, etc.) and SET (THTF test 10 MM, 10DD, etc.).

For the purpose of the optimization of EPR's ECC mode special attention was devoted to phenomena related to countercurrent flow in the hot leg of the main coolant line. Therefore, the code was assessed against UPTF tests 11 and 26.

COCO

COCO models are validated against representative tests, e.g. Batelle-tests.

S-RELAP5/COCO-COUPLING

Comparative plant calculations were performed with both the iterative and the simultaneous method.
3.3 QUALIFICATION AND VERIFICATION (CONT'D.)

HOT LEG FLOW PATTERNS AND CODENSATION EFFICIENCY OF HOT LEG INJECTION DURING REFILL/REFLOOD POST-TEST CALCULATION OF UPTF TEST No. 26 RUN 231

TEST DESCRIPTION - The effect of condensation on counter-current flow in the hot leg was investigated. Strongly subcooled ECC-water was injected into one hot leg at mass flow rates of 150 and 400 kg/s. Steam injected into the core simulator flows from the upper plenum via the hot legs towards the steam generator simulators.

ADRESSED S-RELAP5 MODELS - Interfacial friction, interfacial mass transfer, counter-current flow limitation and flow map.

ASSESSMENT RESULTS

S-RELAP5 accurately predicts thermal-hydraulic phenomena occurring in hot leg and upper plenum under LBLOCA conditions as typically expected in a PWR with hot leg injection. The code systematically underestimates the ECC-efficiency.

Hot leg flow patterns are in general well simulated. However, the code predicts the transition from stratified flow to total water carry-over at lower steam mass flow rates than in the test; this is conservative in terms of core cooling.

Fairly good agreement between calculation and test was found with respect to both the formation of plugs at high injection rates and the intermittent water delivery to the upper plenum. The resulting average condensation efficiency in hot leg and upper plenum is underestimated by 15 - 20 %, which is again penalising for the ECC-efficiency.
BOUNDARY CONDITIONS FOR UPTF TEST 26 RUN 231

- ECC injection
- Steam injection

Water level measurement
Temperature measurement

Test vessel
3.3 QUALIFICATION AND VERIFICATION (CONTD.)

COUNTER-CURRENT FLOW OF STEAM AND SATURATED WATER
POST-TEST CALCULATION OF UPTF TEST No 11

TEST DESCRIPTION - UPTF Test No. 11 is a quasi-steady state, separate effect test designed to investigate the conditions for countercurrent flow of steam and saturated water in the hot leg of a PWR. Countercurrent flow in the hot leg was simulated by venting steam from the primary system through UPTF broken loop hot leg and injecting saturated water into the inlet chamber of steam generator simulator. The test consisted of a series of flow conditions mapping out the countercurrent flow curves at 0.3 and 1.5 MPa.

ADRESSED S-RELAP5 MODELS - Interfacial friction and counter-current flow limitation.

ASSESSMENT RESULTS

Without a CCFL model, the predicted liquid downflow rate suddenly changes from complete delivery to complete carry-over; partial delivery is not calculated. Thus the CCFL option can be used to restrict the flow to a flooding curve defined by a user-supplied correlation.

Using a CCFL correlation of Wallis type with the coefficients \( m = 1.0 \) (slope) and \( C = 0.644 \) (intercept with axis), excellent agreement with test data is achieved.
S-RELAP5 POST-TEST CALCULATION OF UPTF TEST 11

COMPARISON BETWEEN PREDICTED AND MEASURED FLOODING CURVES

![Diagram showing data error bands and flooding curves for different pressures.]

- Data error band
- DATA
- With CCFL
- No CCFL, 15 bar
- No CCFL, 3 bar

p = 15 bar

p = 3 bar

4 ANALYSIS RESULTS

4.1 MAIN INITIAL AND BOUNDARY CONDITIONS

<table>
<thead>
<tr>
<th>DESCRIPTION</th>
<th>EPR</th>
<th>KONVOI</th>
</tr>
</thead>
<tbody>
<tr>
<td></td>
<td>BEST-ESTIMATE</td>
<td>CONSERV.</td>
</tr>
<tr>
<td>Reactor power</td>
<td>100 %</td>
<td>103 %</td>
</tr>
<tr>
<td>Decay heat</td>
<td>DIN + 0σ</td>
<td>DIN + 2σ</td>
</tr>
<tr>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>LHGR</td>
<td></td>
<td></td>
</tr>
<tr>
<td>average value</td>
<td>155 W/cm</td>
<td>160 W/cm</td>
</tr>
<tr>
<td>maximum value</td>
<td>360 W/cm</td>
<td>450 W/cm</td>
</tr>
<tr>
<td>Fuel stored energy</td>
<td>realistic</td>
<td>realistic</td>
</tr>
<tr>
<td>Availability of SIS</td>
<td>no</td>
<td>1 diesel</td>
</tr>
<tr>
<td>maintenance</td>
<td>no</td>
<td>no</td>
</tr>
<tr>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td></td>
<td></td>
<td></td>
</tr>
</tbody>
</table>

a) Depending on break size, most penalizing failure and maintenance assumptions were used for KONVOI-licensing calculations, i.e. for SBLOCA, 2 diesels are considered lost due to SF and maintenance, and for LBLOCA SF of the isolation check-valve connecting the SIS with the hot leg fails to open; additionally, one hot leg accumulator is considered in maintenance (i.e. availability of five of eight accumulators)
4.2 DOUBLE-ENDED GUILLOTINE BREAK

Best-estimate calculations of a 2A cold leg break of the main coolant line with conservative initial and boundary conditions performed for

- EPR + SIS of KONVOI (combined injection)
- EPR + SIS with cold-leg injection
- EPR + SIS with hot-leg injection

- Calculations demonstrate that ECC efficiency of a SIS with 8 accumulators and combined injection is equivalent with that of the SIS with only 4 hot leg accumulators: PCTs and quenching times are practically equal.

- This agrees well with experimental results: PKL tests II B-2 and II B-8 demonstrate that the impact of cold leg accumulators on the end-of-blowdown and reflood phases is negligible.

- For EPR an ECC-mode with hot leg accumulators is from the point of view of ECC-efficiency as good as the SIS of KONVOI type (i.e. combined injection).

- Differences between the ECC-mode with hot leg accumulators and the one with cold leg accumulators are negligible with respect to ECC-efficiency.

- Containment peak pressure shows low sensitivity to the ECC-mode. The peak pressure is in all cases approximately 330 kPa, i.e. much lower than the design pressure of 650 kPa.
EPR with Various Safety Injection Systems
2A Cold Leg Guillotine Break
SHORT TERM EVOLUTION OF CONTAINMENT PRESSURE

LONG TERM EVOLUTION OF CONTAINMENT PRESSURE

EPR with Various Safety Injection Systems
2A Cold Leg Guillotine Break

389
Envelopes of measured cladding temperatures in PKL tests II B-2 (2A cold leg break with combined accumulator injection) and II B-8 (2A cold leg break with hot leg accumulator injection).
4.2 DOUBLE-ENDED GUILLOTINE BREAK (CONT'D.)

LINK TO THE PAST

Results of present analyses may not be understandable, when comparing them to licensing calculations performed 10 or 15 years ago. Present analyses demonstrate that for EPR cold leg or combined injection are practically equivalent with respect to ECC-efficiency and containment peak pressure, whereas calculations performed in the past showed that cold leg injection led to higher PCTs than the combined injection.

Differences between the old and the new calculation results are due to the following facts:

1. Former PWR licensing calculations were performed in accordance with 10 CFR 50.46 Appendix K issued in 1974 and were excessively conservative whereas realistic evaluation models are used for EPR’s safety analyses. These realistic models are based on results of the extensive thermal-hydraulic research performed in the last years (e.g. the UPTF-TRAM program the 2D/3D-Program including UPTF, CCTF and SCTF).

2. Initial and boundary conditions used for EPR safety analyses are different from those used in the past.

3. The maximum LHGR of the fuel rods is ~ 20 % lower in the EPR than in KONVOI.
4.3 OTHER LOCAs

The following scenarios were studied:
- Surge line break (hot leg, 830 cm²)
- SIS line break (cold leg 390 cm²)
- SBLOCA (44 and 80 cm² cold leg leaks)

Calculations have been performed only for the ECC-mode with cold leg injection. The comparison against combined injection is performed by engineering judgment.

THE MAIN EPR SAFETY SYSTEM AND ENGINEERED SAFEGUARD ACTIONS RELEVANT FOR ACCIDENT ANALYSIS

<table>
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<tr>
<th>SIGNAL</th>
<th>CRITERIA</th>
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<tbody>
<tr>
<td>Reactor Trip</td>
<td>Pressure &lt; 13 MPa</td>
</tr>
<tr>
<td>ECC-Signal</td>
<td>Pressure &lt; 11 MPa</td>
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<tr>
<td>RCP-Trip</td>
<td>LOOP or saturation at cold leg</td>
</tr>
<tr>
<td>MHSI</td>
<td>Pressure &lt; 8.0 MPa and ECC-Signal + 30 s</td>
</tr>
<tr>
<td>LHSI</td>
<td>Pressure &lt; 2.0 MPa and ECC-Signal + 30 s</td>
</tr>
<tr>
<td>Partial Cooldown</td>
<td>ECC-Signal</td>
</tr>
<tr>
<td>EFWS</td>
<td>SG level &lt; 8 m + 50 s</td>
</tr>
</tbody>
</table>

a) Delay due to diesel load step.
b) SGs are cooled down via the SG relief valves from 9.15 MPa to 6 MPa with gradient of 100 K/h.
4.3 OTHER LOCA\textsc{ds} (CONT\textsc{d.})

SURGE LINE BREAK

(830 cm$^2$ hot leg break)

Core uncovery caused by core level depression due to "manometer" oscillations. From the point of view of cladding temperature, this is insignificant: the PCT is 370°C.

Accumulator injection did not start at this time. Therefore, the only phase which could be relevant with respect to ECC-efficiency is not controlled by accumulator injection. At 120 s RCS pressure decreases below 4.5 MPa and accumulator injection starts. The amounts of subcooled water injected by accumulators are sufficient to suppress any steam production in the core. As a consequence, RCS pressure drops below 2.0 MPa and LHSI starts. At 240 s a stable state is reached.

The break of largest connecting line to RCS is also with respect to containment response not challenging for the SIS. Maximum containment pressure reached is 320 kPa.

ECC efficiency of cold leg injection and combined injection are equivalent.
ECC-Mode Cold Leg Injection
Pressurizer Surge Line Break (830 cm²)
ECC-Mode Cold Leg Injection
Pressurizer Surge Line Break (830 cm²)
4.3 OTHER LOCAs (CONTD.)

SIS LINE BREAK
(390 cm² cold leg break)

A core heat-up phase begins at 100 s, induced by core level depression. The PCT is 515°C.

Fuel rods are rewet at 270 s by accumulator injection. At 280 s RCS pressure decreases below 2.0 MPa and LHSI begins.

Injection mode is obviously not relevant for SIS line break scenario. Core heat-up is not controlled by SIS; it is caused by core level depression and not by insufficient coolant inventory.

More important for ECC is accumulator pressure. Accumulators with low pressure (i.e. 2.6 MPa) would be slightly more disadvantageous: during RCS depressurization from 4.5 MPa down to 2.6 MPa fuel rod cladding temperatures would increase due to loss of coolant inventory. This would lead to slightly higher PCTs.

Nevertheless, ECC efficiencies of cold leg and combined injection are practically equivalent.
ECC-Mode Cold Leg Injection
SIS Line Break (390 cm$^2$)
4.3 OTHER LOCAs (CONT'D.)

SBLOCA

(44 cm² and 80 cm² cold leg leaks)

Secondary side partial cooldown activated by ECC signal enables RCS depressurization below 8.0 MPa; MHSI can start.

For the 44 cm² leak a short and limited core uncoverage phase caused by core level depression takes place between 900 s and 1060 s. The PCT is 370°C. At 2100 s accumulator injection begins.

The 80 cm² cold leg leak scenario is similar to the 44 cm² cold leg leak from the point of view of thermal-hydraulic phenomena; only timing of various events is shorter. Between 760 and 820 s there is a short and limited core uncoverage phase caused by core level depression. The PCT is 390°C. Accumulator injection begins at 1440 s.

The ECC efficiency is not sensitive to ECC-mode. Core heat-up phases are short and limited (low PCT); they are caused by core level depression and are not controlled by SIS. Accumulator injection is not relevant for emergency core cooling.

It can be concluded, that cold leg and combined injection have equivalent ECC-efficiencies for SBLOCA.
ECC-Mode Cold Leg Injection
SBLOCA - 44 cm² Cold Leg Leak
ECC-Mode Cold Leg Injection
SBLOCA – 80 cm² Cold Leg Leak
5 CONCLUSIONS

EPR’s safety injection system is well balanced: it ensures high ECC efficiency over the whole accident spectrum, it limits loads to containment and has at the same time favorable features for SGTR mitigation (e.g. MHSI shut-off head of 8.0 MPa).

The evolution and the optimization of the EPR’s ECC-mode would have not been possible without using best-estimate analysis methodologies.

The more conservative models, methods and assumptions are, the more difficult it is to optimize systems and to reduce or avoid “over-engineering”.

For the whole LOCA spectrum the ECC efficiency of EPR’s SIS with cold leg injection is practically equivalent to ECC efficiency of a SIS of KONVOI type with combined injection. The smaller the break, the more insignificant are differences between cold leg injection and combined injection.

ECC mode has a negligible impact on containment pressure and temperature evolution during LOCA. Neither with combined injection nor with cold leg injection a containment spray system is needed.
Requirements for BE-Containment Safety Analysis

J. Rohde, B. Schwinges
Gesellschaft für Anlagen- und Reaktorsicherheit (GRS) mbH

Containments of water cooled power reactors traditionally have been designed to withstand the mechanical and thermal consequences of a spectrum of postulated loss of coolant accidents (LOCA). The Design Basis Accident (DBA) scenarios, which cover a range of ruptures in the primary or secondary coolant lines up to the large diameter of the main recirculation line have been internationally adopted to support the specification for the design and the safety assessment of nearly all engineered containment safety systems. For the analysis of the thermal hydraulic processes in containment to determine LOCA induced loads the proceeding is two fold:

To determine the max. accident pressure and intercompartmental pressure differences relevant for the containment lay-out the conservative approach is still valid. A conservative setting of parameters (e.g. model- and input parameters) can clearly be defined. Such conservative analyses could also be performed with sophisticated (BE) containment codes choosing special models by input which will lead to conservative results (e.g. heat transfer correlations) or setting specific input parameters conservatively and adding some safety margins. The choice of conservative settings depends on containment type (dry containment or pressure suppression system) and the safety aspects to be investigated.

But already a small break LOCA event leading to a non-homogeneous energy and steam distribution in the containment and especially when analyzing severe accidents with process dependent scenario development require a BE-analysis. This is due to the strong coupling between the main parameters, the occurrence of more local effects.
and the relevance of long-term distribution processes. A conservative approach could lead to an overestimation of the influence of one parameter against the others and thus lead to wrong results, with a strong impact on severe accident management measures (containment venting, lay-out of hydrogen mitigation systems).

Therefore BE analysis is needed in containment investigations. For such analysis the following requirements have to be considered:

- use of well validated BE code
- appropriate choice of physical models (if possible by input) and model parameter setting (e.g. heat transfer correlations, equilibrium zone model etc.)
- problem oriented, appropriate nodalisation
- input parameters for geometry and system simulations
- complete simulation of all energy and mass sources and sinks, including fission products and its local distribution
- initial thermo-hydraulic conditions (operational)
- material properties

The appropriate choice of the models and correlations should be recommended by the code developer on the basis of the validation work performed. The models are often improved to achieve a better agreement with measured results. Depending on safety oriented aspects or questions to be answered, for the analysis (conservative, BE) different models have to be selected.

The degree of the nodalisation or discretisation depends on the phenomena expected to occur (gas plumes, stratified conditions, non-homogeneous or only homogeneously mixed atmosphere) and which local results or conditions shall be achieved.

How to proceed and which quality of input data is required for BE containment analysis will be explained based on the GRS-code RALOC. But first of all some informations will be given about the status of the code RALOC.

This computer code, developed and used for containment analysis by GRS, is able to evaluate:
- pressure- and temperature build-up and history
- local temperature- and pressure distributions
- energy distribution and local heat transfer and heat conduction in structures
- local gas distributions (steam and different non condensable gases)
- hydrogen combustion and catalytic recombination
- water distribution
- mass- and volume flow for the release of fluids via opening and leakage
- heat- and combustion gas distribution during fires

Calculations can be performed for simple and multi-compartmented containments and closed buildings of nuclear power plants, as well as for compartmentalized systems (buildings, tunnels, pit system) with more or less large openings to the environment. Mainly the consequences of design basis accidents and severe accidents were analyzed using BE assumptions with the code in containment of LWRs i.e. for PWR and BWR, but also in containment of VVER power plants. Some fire events have been investigated, too.

For the description of the physical processes during an accident propagation arbitrary compartment systems and -geometries can be simulated by specified volumes (then so called 'lumped parameter' concept). The conditional changes related to location and time are reduced to a purely time dependent behavior within the control volumes (nodes). These volumes are connected by 'junctions'. For the simulation of heat transfer and heat conduction via walls and internal components specified structures can be coupled to the nodes. The heat conduction is described in one dimension, for the simulation of heat transfer processes (heat- and mass transfer) different models and correlations are available.

For a realistic description of accident sequences the simulation of engineered systems is possible like pumps, heat exchangers, ventilation systems, weir, doors and flaps of different kinds with inertia effect, spray systems, catalytic and thermal recombiners and pressure suppression systems.
For the validation of the different models a large amount of experiments have been analysed, as pre- and post-calculations. The code has successfully used for 4 International Standard Problems and some other benchmark exercises.

Concerning the data required for the simulation of a large dry containment in RALOC the following has to be taken into consideration:

In order to achieve a high degree of predictability, large scale resolution capacity of the input data for the analyses with RALOC are principally required. Especially, this is concerned for the representation of real compartments of a reference plant by the so called model compartments of the code. The high degree of detailing is required for the simulation of long time convection processes in a containment which consequently determines e.g. the resulting local temperatures or gas concentrations needed for the conception of Severe Accident Management (SAM-Measures).

Also a high degree of detailing is required to cover the junction openings between adjacent compartments according to sizes, places, positions and direction. These junction openings can be splitted in free openings, openings resulting from pressure differences like doors, burst flaps, rupture discs as well as in ventilation and drainage junctions.

The exact knowledge of the real containment is required for this. These detailed knowledges are gained from the drawings of the building, supplementary documents of the vendor and utilities as well as from plant inspections. As for example, the rupture discs made of very thin stainless steel and located above the steam generators in German PWR containments are represented exactly by a distribution function of pressure difference of the opening. Such rupture discs create according to ventilation technique a separation between the passable compartments during operation and the non-passable area. Thus, they function as barriers for convection flows in the containment during an accident. The high data quality is also asked for other openings, dependent on the build-up of pressure differences, for which one applies realistic values of pressure differences generating openings as input.

Also for free volumes of individual compartments as well as for the heat conducting structures within them, and for the ceiling and walls of the rooms made of concrete, steel and other materials, only the most realistic values are to be applied.
Out of the above mentioned requirements for the preparation of the input-data, a RALOC nodalisation was developed for a German PWR containment for SAM investigations, which consists of 106 compartments for the containment vessel, 14 for the outer annulus section and 9 for the auxiliary building. Moreover, the input dataset for RALOC contain appr. 470 junctions and 240 heat slabs.

A BE nodalisation has to take into account the accident sequence to be investigated and for the release of energies and masses the source size, strength and release location with respect to the position in the containment building (large LOCA gives more homogeneous distribution of energy and gases). The more detailed the nodalisation is chosen the more detailed description of the distribution processes is achieved, which leads to more realistic description of the processes. For simplifications of the nodalisation a nodalisation study is needed in order to prove that the reduced number of nodes does not influence the phenomenological behavior to be investigated. The code developer should provide nodalisation rules and recommendations for the user.

The initial conditions of the atmosphere in the containment especially humidity and temperature and possible leakages to the environment determine the mass of air available for reactions. The energy storage capacity of the structures is influenced by the initial structure temperatures.

The results of a BE analysis need the specification of the uncertainties as a measure for the credibility. An uncertainty analysis related to the specific investigation should be performed. Different tools for uncertainty analysis are available. Such an uncertainty analysis gives the uncertainty limits or intervals of calculated results representing the combined influences of all parameters involved and sensitivity measures about relative influences of parameters on the calculated results like regression coefficients, correlation ratios and confidence interval.

The specific requirements to perform containment BE analysis will be underlined by examples, given in the presentation.
Requirements for BE-Containment Safety Analysis

Content

- Principal ideas to methodology to assess analysis accuracy

- Methodology for DBA containment assessment
  - Code Validation
  - Sensitivity Studies
  - Conservative Calculation

- Severe Accident (BDBA) analysis of reactor containments
  - Validation of computer codes
  - Uncertainty of calculated results
  - Method to quantify uncertainties
  - Specific requirements for BE-containment safety analyses

- Some Conclusions

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Principal Ideas to Methodology to Assess Analysis Accuracy

- For safe operation of reactors a design which could withstand max. accident load conditions is needed
- Calculated accident loads still have uncertainties
- To overcome existing uncertainties: conservative calculations + safety margins are used
- A conservative hypotheses may not always lead to conservative results
- For the analysis of the primary system behavior during an accident more than one strongly coupled safety parameter exist. This makes it nearly impossible to clearly identify a conservative setting of input parameters.

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Principal Ideas to Methodology to Assess Analysis Accuracy

- Uncertainty analysis (e.g. GRS methodology SUSA) was therefore developed
- For containment analysis the situation is somewhat different

Containment loads

DBA:
conservative def. clear:
what gives max. loads
(P, T)
Parameters are in general not coupled

BDBA, severe accidents:
conservative parameter can lead to wrong results and conclusions (sequence)
Methodology for DBA Containment Assessment

- Conservative calculations can be done with
  - conservative codes, i.e. model correlations (i.e. heat transfer etc.) + safety margin
  - best estimate codes with conservative setting of input parameters + safety margin

- Conservative setting of input parameters depends on
  - containment system (dry containment, bubble condenser)
  - aspects of assessment

- Validation of used codes on an experimental base
  - at least giving a qualitative agreement (characteristics)
  - not too large deviations from measured data (parameter dependent, i.e. \( p(t) \pm 5 \pm 10\% \))
  - prove of describing scaling effect by validation on test facilities of different sizes
Methodology for DBA Containment Assessment
(cont.)

- Conservative setting of input parameters can clearly be defined based on sensitivity studies
- Parameters of interest are decoupled
- Sensitivity studies to be performed for
  - model parameters
  - system/geometry parameters
  - material properties
  - initial TH (operational) conditions
- Nodalisation studies
- Max. (peak) accident pressure/temperature
  - full pressure, dry containment
    - main parameters are addressed in the guidelines for max. pressure
    - for small breaks pressure development is dependent on break location
      (need for a detailed nodalisation)
Validation of Computer Codes

Integral and separate effect tests

Initial conditions
Boundary conditions
Measurement errors
Model approximations
Material properties
Numerical algorithms
Nodalisation

Experimental data

Calculated results

Accuracy
Validation of Computer Codes (cont.)

- Selection of individual tests for a complete validation
  
  ♦ Each phenomenon should be addressed in test facilities of different scale, to check the code ability of transferring to another scale
  
  ♦ Each separate model in the code should be validated on separate effect tests, if available in different scale, 
    → to prove the validity of the model for larger scale (in integral experiments) 
    → thus avoiding compensating errors in validation based on integral tests

- Experiences from validation
  
  ♦ Thermal-hydraulic containment codes approximate the physical behavior with more or less accuracy#
  
  ♦ Agreement of calculated results with experimental data often obtained by changing model parameters

Validation experience is the basis to quantify parameter uncertainties
Initial and Boundary Conditions for the Determination of the Maximum Pressure Build Up (PWR)

- Size and location of the break area (parametric study)
- Mass and heat transfer of the secondary side of one steam generator (SG)
- Additional energies out of:
  - Core region
    (Decay heat and residual heat)
  - Residual heat from the metallic structures of the primary system
  - Heat transfer from the SG to the primary side
- Structures to be considered
  - Volume of the primary and secondary system: +2 %
  - Net volume of the containment: -2 %
  - Surfaces of the structures of the containment with internals
    Steel of the containment: -2 %, Internal structures: -10 %
    maximum thickness of the decontamination painting
Initial and Boundary Conditions for the Determination of the Maximum Pressure Build Up (PWR) (Cont.)

- Initial conditions within the containment
  - Temperature and humidity low
  - Starting pressure high

- Heat transfer correlation
  (conservative: Tagama / Uchida / Slaughterbeck)

- Thermodynamic nonequilibrium between water-steam-phase
  (or by calculation in thermodynamic equilibrium: + 0.3 bars)

- Safety margin of 15 % in addition to the calculated maximum pressure
  (Nominal load state, calculation and model uncertainties)
Maximum Pressure in the Containment due to the Break Size (PWR)
Influence of Additional Steel Structures for the Pressure Course
Influence of the Addition of One Secondary Steam Generator for the Pressure Course
Maximum Pressure Large Dry Containment
(PWR)

- Free volume
  about 70,000 m³ (1300 MWe)

- Pressure peak at 15 - 20 s

- Influencing parameters:

<table>
<thead>
<tr>
<th>Parameters</th>
<th>Part of the pressure build up, %</th>
</tr>
</thead>
<tbody>
<tr>
<td>Integral energy (Initial conditions: normal loading condition at the end of pressure release of the RPV)</td>
<td>70</td>
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<tr>
<td>Secondary energy input</td>
<td>10</td>
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<tr>
<td>Structures not taken into account</td>
<td>10</td>
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<td>Thermodynamic nonequilibrium water-steam</td>
<td>5</td>
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<tr>
<td>Manufacturing tolerances</td>
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<tr>
<td>Decay and residual heat from structure materials</td>
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<td>Break size</td>
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<td>Safety margin by RSK-Guidelines</td>
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Uncertainty Analysis for Accuracy of Severe Accident Analysis of Reactor Containments

- Severe accident analysis is carried out by computer simulation with complex system codes

- Best estimate codes used with conservative parameters can lead to wrong sequences

- A conservative hypotheses may not lead to really conservative results

- Replace evaluation model by best estimate calculations + quantification of remaining uncertainties

- Quantification of uncertainty in code applications
  - reveals major sources of uncertainty of calculated results
  - enables to determine margins to safety limits
  - guides further code development
  - prioritizes experimental investigations

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Comparison of RALOC-Results for One-Node and Multi-Node Configuration

**Pressure**

- Multi-Node Conf.
- One-Node

**Temperature**

- Multi-Node Conf.
- One-Node

**Legend**

- Node 1
- Node 2
- Node 3
- Node 4
- Node 5
- Node 6
- Node 7
- Node 8
Comparison of RALOC-Results for One-Node and Multi-Node Configuration
Effect of the Degree of Accuracy of Containment Nodalisation on Hydrogen Combustion during a Severe Accident Sequence

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Requirements on the RALOC Nodalization for a Real Plant to Perform Investigations on Gasdistribution and Hydrogen Recombination

- A high degree specification is required concerning the nodalization of a containment to simulate
  - longterm convection pattern in the containment
  - local positioning of catalytic recombiners
  - release areas for mass and energy into the containment (e.g. break location)
  - existing flow connections between the different compartments

- Consideration of all types of flow-connections between the compartments (size, location, direction) e.g.
  - free existing openings
  - openings, being developed from pressure differences like doors, flaps, rupture foils
  - connections via the ventilation system
  - drainage flow paths

- Plant specific data on
  - normal plant operating conditions for the containment atmosphere (local pressure, temperature and humidity)
  - free volumes of the compartments
  - realistic detailed registration of all structures and internals in the different compartments functioning as heat sinks and sources
Nodalisation of a large dry containment to analyze a severe accident sequence in a German 1300 MW PWR:

- zones

<table>
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$\Sigma$ 461

- junctions

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<tbody>
<tr>
<td>free atm. incl. leakages and fan system</td>
<td>197</td>
</tr>
<tr>
<td>drainage</td>
<td>127</td>
</tr>
<tr>
<td>variable openings*</td>
<td>137</td>
</tr>
<tr>
<td>(*doors, rupt. foils, rupt. plates, flaps)</td>
<td></td>
</tr>
</tbody>
</table>

- catalytic recombiners

$\Sigma$ 53

- structures

$\Sigma$ 226

- input tables for mass and energy release into the containment (MELCOR results)

<table>
<thead>
<tr>
<th>break position</th>
<th>distributed</th>
<th>cavity</th>
</tr>
</thead>
<tbody>
<tr>
<td>water steam $H_2$</td>
<td>FP-decay heat, volatile FP-decay heat, depos. system heat (stored energy)</td>
<td>water steam $H_2$ CO$_2$ CO system heat heat from molten pool</td>
</tr>
</tbody>
</table>

428
Effect of the Degree of Detailness for the Dome-Nodalisation on Local Hydrogen Concentration
Severe Accident LP-Sequence in a German PWR-Plant with working PARs

(Passive Autocatalytic Recombiner)

Ankara, Turkey  •  June 29 - July 1, 1998
Calculated Normalized Energy Distribution During the First Phase of a Severe Accident (LP-Scenario) in a Typical German PWR
Validation of the RALOC Code:

Post Calculation for the Test HDR E11.4, Comparison of Local Temperatures and Gas-Concentrations in the Dome Area (elevation: 34.8 m)

Ankara, Turkey  •  June 29 - July 1, 1998
Objectives of the Probabilistic Approach (SUSA)

- Support to the user to analyse best estimate code results using a global approach

- Analyse uncertainties on both
  - code modeling and
  - input parameters

- Evaluate the contribution of each input parameter to the overall uncertainty
Some Conclusions

Results of uncertainty analysis for validation taken from:

Comparison of calculated uncertainty range with measured range of experiment

- Uncertainty analysis for integral experiment
- Check whether experimental results are bounded by uncertainty statement
- If not, revise list of input uncertainties and their specified uncertainty ranges
- If yes, uncertainty analysis to the plant
  → status of plant
  → scaling
- Analysis of scatter plots in the case of unexplained differences between sensitivity measures

Ankara, Turkey • June 29 - July 1, 1998
Some Conclusions (cont.)

- Statistic analysis of uncertainties is a global approach
- Applications demonstrate the feasibility of the method
- Difficulties for evaluating the probabilistic characteristics of the uncertain input parameters
  - probability distributions: in most cases not available
  - uncertainty ranges: estimation based on physical data
  - degree of dependence: to be estimated
- Output parameters should be selected carefully
  - to reach the objectives from the statistic analysis
  - to be sure that the code was used in its validated area
    (additional responses if needed)
OVERVIEW OF UNCERTAINTY ISSUES AND METHODOLOGIES

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Glaeser H. (GRS, Munich, G)
Lage C. (ENUSA, Madrid, E)
Wickett T. (AEAT, Winfrith, UK)

ABSTRACT

The present paper deals with the problem of uncertainty evaluation from the predictions of thermalhydraulic best estimate codes adopted in nuclear reactor technology. The origins of the problem are discussed together with the necessity. The attention is focused toward the description of the main features of five uncertainty methods proposed by OECD (Organization for Economic Cooperation and Development) countries, recently subjected to deep scrutiny in the framework of an activity promoted by the CSNI (Committee on the Safety of Nuclear Installations).
1. INTRODUCTION

Uncertainty evaluation in nuclear reactor thermalhydraulics is necessary for quantifying reactor safety margins in predicting transient scenario progressions in Nuclear Power Plants.

In September 1988, following the improved understanding of ECCS (Emergency Core Cooling System) performance during various reactor transients, the NRC reviewed and amended the requirements previously fixed in the 10CFR50 (§ 50.46 and App. K). As a consequence of the conservatism of the methods specified in App. K, being responsible of reactor operating restrictions and according to industry requests for improved evaluation models, the following approach was oriented toward basing the licensing decisions on realistic or so-called, Best-Estimate (BE) calculations of plant behaviour. The amendments essentially concern the permission of using BE models for ECC performance calculation as an alternative to codes (Evaluation Models) that use the App. K conservative requirements. The rule changes also include the quantification of uncertainties of best estimate calculation.

Thermalhydraulic system code calculations are affected by errors arising from several causes, including the unavoidable approximations in the constitutive equations, from the limited capabilities of numerical solution methods, from uncertainties in the knowledge of boundary and initial conditions, from errors in setting up the nodalization. These can be characterized by hundreds of parameters that are typically part of the input deck for a system code calculation suitable for predicting a transient scenario in a Nuclear Power Plant. This happens notwithstanding the high code performance level and the systematic qualification processes, nowadays in progress or completed. It is necessary to remind, in this connection, that the user choices strongly affect the code results, through the so called "user effect", ref. /1/.

Following some pioneering work, promoted by the US Nuclear Regulatory Commission and leading to the proposal of the CSAU (Code Scaling, Applicability and Uncertainty), e.g. refs. /2/ to /4/, different uncertainty methodologies have been developed by different research organisations, e.g. refs. /5/ and /6/ in order to evaluate the reliability of any thermalhydraulic code calculation, taking into account the possible sources of error.

Essentially, two different and independent approaches have been proposed: CSAU and the uncertainty methodologies proposed by AEA/GRS and IPSN (e.g. ref. /6/) consists in trying to identify the main sources of uncertainty among the code input parameters, also by means of engineering judgement, in analysing their effects upon the calculation results; the other approach, followed in the UMAE, does not care directly on the sources of uncertainty in the input parameters but only on their effects upon the results, having as hypothesis that the error made by code is of the same order of magnitude when predicting a nuclear reactor transient and the experiments performed in properly scaled Integral Test Facilities, ref. /7/. Inside the first approach, methods fully based upon the statistics (e.g. the GRS method) and methods only relying on deterministic procedures (e.g. AEA method) have been distinguished. The approach followed by UMAE does not allow, directly, the identification or the ranking of the origin of the errors; in addition, all sources of uncertainty are considered, not in the form of ranges of variations of the input parameters but in the form of errors in the output parameters. The need of experiments in the UMAE, prevents or limits its use to cases where experimental data are available.

The purpose of the present paper is to show the origin of the
uncertainties in system codes predictions also giving an idea of the goals of uncertainty studies. In addition, five methodologies recently proposed by AEAT (Winfirth, UK), ENUSA (Madrid, Spain), IPSN (Cadarache, France), GRS (Munich, Germany) and University of Pisa (Pisa, Italy) and subjected to deep mutual scrutiny by developing teams and to independent peer review, are described.

2. REASONS AND OBJECTIVES FOR UNCERTAINTY STUDIES

Uncertainty evaluation constitutes a complex task that may need large resources. Therefore the origins or the reasons for carrying out the analyses must be clarified together with the objectives or the expected benefits.

2.1 Reasons and origins

The system thermalhydraulic codes, whose results are subject to uncertainty evaluations, are far from being perfect; their solutions are approximate, the amount of approximation constituting the goal of uncertainty studies.

The characterization and/or the origin of the approximations are summarized hereafter. Reference is made to the use or the features of two system codes (Relap5 and Cathare) that are in use at DCMN (Department of Mechanical and Nuclear Constructions) of University of Pisa, but can be extended to other codes.

A thermalhydraulic system code deals with the solution of balance equations based upon first principles of physics supplemented by empirical correlations; these are numerically solved utilizing boundary and initial conditions supplied by code user. The code itself generally consists of more than $10^7$ fortran statements and the input decks may reach $10^4$ 'cards' (order of magnitudes for both cases): errors can easily be part of such packages of electronic statements.

An indicative list of approximations is given hereafter, making reference to steam-liquid mixtures.

A) Balance or conservation equations are approximate. Not all phenomena that occur when liquid and steam interact, are directly considered by the conservation (no matter what is the importance of the phenomenon): e.g. droplets interactions may create larger droplets moving at a velocity different from the original droplets, changing system energy and momentum. The equations are solved within cylindrical pipes without geometric discontinuity: situation not frequent in NPP, lacking info needed from user.

B) Presence of different fields of the same phase. Liquid in a two phase mixture may be present in the form of droplets or film (e.g. two fields) that are characterized by different velocities, temperatures, etc.: only one velocity, temperature, etc., is calculated by the current codes.

C) Geometry averaging at cross-section scale. Different velocity profiles happen in the reality depending upon local values of thermodynamic quantities and upon history: averaged values are provided by codes that are valid in a limited number of situations.

D) Geometry averaging at volume scale. One velocity is associated to a volume or hydraulic mesh in the main fluid direction, i.e. assumed axis for
motion. Let us consider the simulation of the lower plenum of a PWR at steady state nominal operating conditions. Liquid coming from downcomer is characterized by a spectrum of velocity vectors downward oriented that change within the space of few meters: the new spectrum is characterized at the core entrance area by upward oriented velocity vectors. In the code simulation, the entire process involves two velocity values (not two spectra) and occurs in one "geometrical point": the node(s) simulating the downcomer and the core entrance region have a single downward and upward velocity, respectively; the velocity change happens in the connection between the two nodes. The simulation is much more complex when two phases are involved in the above geometry.

E) Large and small vortex or eddy simulation. In single and two-phase flow, unavoidably vortices and eddies appear with different modalities depending upon the geometric and thermodynamic boundary conditions; these create energy and momentum dissipation not directly accounted for by codes.

F) The second principle of thermodynamics. This is not necessarily fulfilled by the code solved equations, that basically deal with the simulation of irreversible phenomena. During the simulation of an irreversible process inside an isolated system, entropy may not be predicted to increase. The amount of error in predicting relevant quantities is strongly a function of the application.

G) Numerical solution is approximate. The approximate balance equations are approximately solved by special numerical methods; the saving of computer memory and timing is a goal for such methods. The amount of approximation involved is not documented in the general case (the mass error constitutes only one example).

H) Correlations implementation and range of validity. Interaction between the phases and between each phase and the walls are simulated by constitutive terms that appear as correlations. Almost all of these come from experiments and, as such, are characterized by ranges of validity; uncertainty in code results originated by correlations are:
- ranges of validity may be not fully specified (pressure, fluid an wall temperature, velocities, void fractions, walls material properties, etc.);
- the correlations may be unavoidably used outside their range of validity (see also the item below);
- the correlations may be approximately implemented into the code due to the needs to fit with other correlations, with the selected unknowns or with the numerical solution scheme; an example of this is reported as Fig. 1 related to the implementation of the Shah correlation for condensation heat transfer coefficient in the Cathare code, as taken from ref. /8/;
- inherent scatter and error in the experimental results on which the correlation is based.

I) 'Steady State' and 'Fully Developed' (SS & FD) flow approximations. All qualified correlations (exceptions are present) must be developed under SS & FD conditions. Unfortunately, almost in no region of the primary loop, during a transient, such conditions apply: unknown influence upon the code prediction results from this approximation.

J) State and material properties are approximate. The solution of the
balance equations implies the knowledge of state and material properties, respectively, for the working fluid and the wall materials. Approximations are, again, unavoidable and produce unknown effects upon results.

K) Code user effect. Code users may interact at different levels with code results, as pointed out in ref. /1/. In principle two or more groups of users having available the same code and the same information for developing a nodalization should arrive at the same results: this is not usually true and differences between results may be attributed to the 'user effect'. The difference may be due to users making different choices within what is allowed by the code manual or errors (choices outside what is allowed by manuals). An attempt to quantify the user effect has been made elsewhere, e.g. ref. /1/: user effect is important when developing a nodalization (i), when interpreting supplied information (ii), when accepting a steady state performance of the nodalization (iii), and when interpreting transient results (iv). Typical effects dealing with items (i) to (iv) are documented in Tab. 1 and Figs. 2 to 4, respectively.

L) Computer/compiler effects. A code installed in any computer machine should produce the same results provided a unique input deck is adopted. This is not the case due to a number of reasons connected with the precision of the machine and with the compiler design, e.g. see refs. /9/ and /10/. From the last one, the Tab. 2 is taken: with reference to the International Standard Problem 26 (OECD/CSNI ISP 26), the same input deck, suitable for Relap5/mod2, run on different computers produced different results at 240 s into the transient. More recent computer-compilers and code releases did not improve the situation also owing to a greater number of compilers actually used.

M) Nodalization effects. The nodalization of a complex geometric/thermalhydraulic system is the result of a wide range brainstorming process from the code user. This starts from the boundary conditions and hardware data and goes through the user expertise and the code guidelines. A number of code input values might not be covered by any logical step, or the user expertise may be inadequate. An example of this situation is given in Figs. 5a) and 5b): this deals with the nodalization of the Lobi facility, refs. /11/ and /12/, in particular, of the connection between Cold Legs and Downcomer. The mass inventory as a function of time is reported as obtained from the two nodalizations identified as a) and b) in Fig. 5a): in case a), the steam produced in the core is predicted to easily flow toward the downcomer and to the break; in case b), more liquid is predicted to flow out of the break, causing larger coolant loss. The comparison with experimental data in Fig. 5b) gives an idea of the effect of this choice. It may be noted that the elevation of the concerned junction is the same in the two cases and that no user guideline might cover this situation: in different experiments, situation b) may be preferable.

N) Imperfect knowledge of Boundary or Initial Conditions (BIC). Boundary and initial conditions affect the transient evolution in the way fixed by the code equations. The problem occurs when the BIC values are unknown or are known with some error. The example shown in Figs. 6a) and 6b), ref. /13/, comes from the simulation of a small break LOCA experiment in Lobi facility. The Time of Accumulator Injection Start (TAIS) and the Peak Cladding Temperature (PCT) are largely affected by the heat losses from steam generator walls to the environment: the range of heat losses, varied
within the limits identified by experimentalists, 30 Kw, causes around 200 K and 200 s output uncertainties in terms of PCT and TAIS, respectively.

0) Code model deficiencies or inadequacies. Code deficiencies cannot be excluded in the most recent code versions made available. One inadequacy was found while analyzing the experiment. Loss of Feedwater with delayed actuation of auxiliary feedwater in Lobi facility, ref. /14/. The auxiliary feedwater was injected in a voided Steam Generator (SG) downcomer at a late time during the transient when dryout occurred in the core rods. During the experiment, the injected liquid partly cooled the SG walls before reaching the bottom of the U-tubes, thus becoming effective for removing heat from the primary side. In the calculation, the injected liquid instantaneously reached the bottom of the SG not rewetting the SG walls when flowing down, since the liquid downflow is assumed to flow down over the whole cross section of the downcomer volume and not as a liquid film as happened in this specific test facility. As a result, void collapse in the primary side was fastly predicted by the calculation but did not occur during the test; consequently, core rewet caused by pressurizer draining was predicted but not measured. This is shown in Figs. 7 and 8; only experimental data are given in Fig. 7: it can be seen that shut-off of electric power was necessary during the experiment to stop surface temperature rise.

2.2 Objectives

As already mentioned, the goal of uncertainty studies is the quantification of the error that is associated with the results of thermalhydraulic system code calculations. A BE code without uncertainty evaluation constitutes an uncomplete work and is useless for practical purposes.

Practical purposes or objectives are identified in the following. BE code results including the uncertainty bands are reported as BEU, hereafter.

* Licensing process. A BE code should erode the conservatism of previous generation Evaluation Models. In other words, making reference to the classical hot rod surface temperature trend versus time, the situation should be as follows: BEU should be lower than the Evaluation model result. and stay well below the allowed limit. Positive consequences for the industry could be:
  - relaxation of current requirements, e.g. one Low Pressure Injection Pump may reveal sufficient instead of two or design requirements (head and flow) for two pumps may be relaxed, with advantages in maintenance or initial system cost, respectively;
  - possibility of uprating power of the plant;
  - use of a unique code for design, maintenance and licensing; the team using the conservative code to fulfill requirements of the licensing authority is not needed.

* Safety analyses. Safety is clearly connected with licensing; nevertheless, safety analyses can be conducted outside the licensing process. An example is constituted by the safety evaluations of existing reactors of Soviet origin, e.g. ref. /15/. A versatile, qualified, and publically available tool must be used to this aim; a BE code is the only applicable tool.
* Design of new plants. The design of the majority of existing plant, at least in relation to the main hardware features, was completed in the 60's without the help of existing codes. Nevertheless BE codes have been used for design confirmation. Envisaged use in the area is:
  - design optimization: e.g. number of U-tubes of SG, position and number of recirculation pumps, volume of pressurizer and of the vessel downcomer, etc.;
  - design of 'passive' reactors: effectiveness of emergency systems is more difficult to demonstrate than in current generation reactors; the use of BE codes appears mandatory.

* Optimization of Emergency Operating Procedures (EOP). Findings from the operation of experimental facilities, from the results of BE codes, from Accident Management and Probabilistic Safety Assessment (PSA) related studies, opened new possibilities in this area. Achievement of BEU may reveal mandatory for demonstrating the suitability and the applicability of any new EOP.

* Operator training and simulators qualification. The training of operators, the part done through plant simulators, needs realistic accident scenarios and NPP feedback to operator interventions. This can only be achieved through BE codes that must be used to benchmark the simpler codes at the basis of the simulator in all conditions of interest. Existing post-processors, including advanced graphical user interfaces allow the direct use of BE codes for operator training.

3. UNCERTAINTY METHODOLOGIES - BASES

When developing an uncertainty methodology, the items A) to O) in sect. 2.1 should be considered as well as the objectives identified in sect. 2.2.

Uncertainty origins A) to J) are embedded into the code and cannot be avoided; on the contrary, the effects upon the calculation results from uncertainty origins K) to M) can be reduced by procedures to ensure quality and eliminate errors, for example pre-processors, training of code users and qualification of nodalizations so its is consistent with code manuals requirements, e.g. refs. /16/ to /18/. Finally, uncertainty origins N) and O) should be carefully considered; rigorous code assessment programs and expertise in developing nodalizations can be helpful in this connection.

3.1 Previous reviews of uncertainty methods

The degree of interest in the uncertainty methods has led to a number of reviews being published recently, refs. /5/, /6/ and /19/ to /22/.

Eight methods dealing with uncertainty are discussed in ref. /5/ together with the principles for identification of uncertainties relevant to thermalhydraulic transient analysis. Holmstrom reviewed CSAU, ABAT and GRS method status in ref. /19/ and in the follow up paper, ref. /20/, described recent developments of those methods and of IPSN and UNAR. Forge, ref. /21/, in the framework of an European Commission activity, published a review describing the CSAU, the ABAT (referred to as the AEAW method), a method proposed by General Electric and the GRS method.
Main findings from refs. /6/ and /22/ are discussed in the next paper (chapt. 4).

3.2 Bases of uncertainty methods

The attention is focused hereafter toward the methodologies that were submitted to a deep review process in the frame of the UMS (Uncertainty Method Study) promoted by the CSNI. As already mentioned, these have been proposed by AEAT (Winfrith, UK), ENUSA (Madrid, Spain), IPSN (Cadarache, France), GRS (Munich, Germany) and University of Pisa (Pisa, Italy) and are identified in the following as AEAT, ENUSA, IPSN, GRS and UMAE (Uncertainty Methodology based on Accuracy Extrapolation), respectively. A full description can be found in the Vol. II of ref. /22/.

A common aspect for all the methodologies is constituted by the use of experimental data in the process; however, modalities in the use and type of needed data are different.

Basically, three approaches have been distinguished.

1) Purely deterministic:

Any use of statistics is avoided; results of various steps of the methodology are checked and evaluated by the methodology user. The most influencing parameters for an assigned transient are selected together with their ranges of variations. The process must end up with a limited number of parameters in order to limit the number of code runs. Input parameters are modified that can be accessible or not to code user (e.g. coefficients of correlations embedded into the code can be modified).

Method(s): AEAT;

Advantage(s): the result of each step must be accepted (thus it is qualified depending upon the expertise involved) by the methodology user;

Disadvantage(s): expertise not always available; expertise may fail; conservative ranges may generate large uncertainty range.

2) Purely statistics:

Any set of input parameters, i.e. including whatever large number, can be processed. The methodology user must characterize the range and the distribution of variation of each parameter. In the case of nodalization changes, these must be defined by the user. The number of performed code runs necessary to get uncertainty bounds is a function of the level of confidence in the results.

Method(s): GRS, IPSN, ENUSA;

Advantage(s): no limitation on the number of considered input uncertain parameters; parameters mostly affecting the uncertain results are identified and characterized;

Disadvantage(s): conservative ranges of input parameters may generate large uncertainty results; expertise still needed to fix ranges of variations of input parameters.

3) Based upon the propagation of code output errors:

Inaccuracies of calculations are characterised by comparing measured and calculated time trends of relevant variables. Relevant variables are measured in Integral Test Facilities designed according to defined principles. Inaccuracies of calculations are propagated from the facilities to the reference system: extrapolation of accuracy to get uncertainty. The basic assumption is that relevant experimental data are
available and include almost all the uncertainty sources expected in the reference transient; other uncertainty sources must be considered separately through proper biases.

Method(s): UMAE;
Advantage(s): experts' judgement minimized or avoided when using the methodology;
Disadvantage(s): the origin of uncertainty does not appear from the results; the concept of extrapolation of accuracy cannot be demonstrated theoretically (however demonstrations of validity can be supplied); range of application limited by the experimental data base.

4. UNCERTAINTY METHODOLOGIES DESCRIPTION AND COMPARISON

The five considered methods (sect. 3.2) are described into detail in ref. /22/ (UMS report) and in refs. /23/ to /27/, related to AEAT, UMAE, GRS, IPSN and ENUSA, respectively. A short description is provided hereafter; MU identifies the Methodology User.

4.1 Outline of the five uncertainty methods

AEAT

The uncertainty statements or input uncertainty ranges should be in the form of "reasonable uncertainty ranges". Such a range is defined as "the smallest range of values (of a given quantity) that includes all values for which there is reasonable certainty that they are consistent with all available evidence".

The method is planned to select input uncertain parameters and ranges that are consistent with the above statement.

Once a qualified code is made available from developers, the code and input uncertainty ranges are used by MU to predict independent data; these must be representative of the processes expected to occur in the plant transient.

If the predicted uncertainty ranges bound the selected independent experimental data, the code and the uncertainty analysis can be used for plant calculations. If not, then further development is needed for the code and/or the input uncertainty data, and the data from the old independent data base move into the development data base. Different independent data would be needed for a subsequent test.

UMAE

The basic idea of UMAE is the use of the accuracy from the comparison between measured and calculated trends of relevant experiments and calculations, respectively. The experiments must come from relevant facilities and the calculation results from qualified codes and nodalizations. This avoids the need to select input uncertainties; also, resulting uncertainty ranges are coming from the process and do not need subjective evaluations.

Development of suitable nodalizations and qualification at the 'steady state' and the 'on-transient' levels are needed; this is true in relation to the considered facilities and plant. Qualitative and quantitative criteria have been proposed to show the related quality level.
The process of nodalization qualification is fully independent from the process aiming at the derivation of the extrapolated accuracy (different data bases are used).

The fulfillment of various conditions (quality of data base, of NPP nodalization, of code performance) allows the finalization of the process, that, vice versa, can be interrupted at different stages. In the first situation, accuracy, coming from several comparisons between measured and calculated trends can be "extrapolated" and becomes uncertainty. This is superimposed to the unique best-estimate code run performed by a qualified NPP nodalization.

**GRS and IPSN**

These two methods are basically the same the only difference being the type of input parameters that are selected in the process.

The methods have the capability to consider the effect of uncertainty of input parameters like computer code models, initial and boundary conditions, other application specific input data and solution algorithms on the calculation results. They are based on well established concepts and tools from probability calculus and statistics.

The selection of an Integral Test Facility experiment addressing the scenario of interest makes it possible to take decisions on dominant phenomena and corresponding computer code models. All model parameters and initial and boundary conditions which potentially contribute to the uncertainty for the chosen test are selected.

Subjective probability distributions or probability density functions are specified for each identified uncertain parameter. If model parameters have contributors to their uncertainty in common, the respective states of knowledge are dependent. This dependence needs to be quantified if judged of importance.

A random value for each uncertain parameter is selected according to the specified subjective probability distribution. The minimum number of code calculations is given by the Wilks formula: this depends only on the chosen tolerance limits or intervals and not on the number of uncertainty parameters considered.

Sensitivity measures are derived in relation to each input parameter. Calculated uncertainty limits are compared with measured integral test data: if they bound the data, they can be used for plant calculations.

**ENUSA**

ENUSA method is based on the CSAU. The PIRT (Phenomena Identification and Ranking Table) process is used to select a reasonable number of input uncertain parameters, i.e. the same objective as in the AEAT method.

The ranges of variations are fixed, again utilizing the same or a similar approach as AEAT but, in addition, SPDF are identified by ENUSA method (in a similar way as GRS and IPSN methods) as opposite to the AEAT method where only the ranges of output variables are considered.

In order to minimize the number of calculation runs, the process of combination of input uncertainties is basically the same as adopted by GRS and IPSN. This makes the difference between the ENUSA method and the CSAU and may justify considering this method as statistically based.
4.2 Comparison among the five uncertainty methods

The common features to the five considered uncertainty methods are the following:
* each method has the capability to calculate error ranges as a function of time, i.e continuous error bands, that bound best estimate code calculation results;
* each method consists of a limited number of main steps and assumptions that appear evident when the method is applied, see also ref. /28/;
* each method requires resources of the order of man-years to be used the first time by a competent (in thermalhydraulics) technician who is unaware either of the method or of the field of application;
* some features of each method are directly connected with the adopted code. This is valid to a different extent in the various cases: examples are criteria for developing nodalisations, or selection of input parameters for uncertainty that may not exist in each code;
* each method requires the selection of a code and of a transient scenario (reference scenario, and reference NPP);
* each method, as already mentioned, makes use of experimental data to a different extent;
* each method needs a qualified code;
* each method aims at providing information useful to decision makers.

The flow diagrams and the adopted nomenclature, e.g. for identifying assumptions, are not uniform or consistent among the methods: as an example, the PIRT is used by CSAU, and then by ENUSA, to screen the phenomena; phenomena screening is also necessary in the AEAT method, but PIRT is not used or even not mentioned. This together with the second starred item above, had to be overcome in the comparison among the methods, as summarised in sect. 3.8 of ref. /22/.

Nevertheless, making use of items proposed in ref. /6/, an additional evaluation of the selected uncertainty methods is performed, Tabs. 3 and 4.

In relation to Tab. 3:

The increase in the number of input uncertain parameters, item 1), causes substantial increases in the needed computational resources for applying AEAT and CSAU: for this reason this number must be minimized. This is not the case for the GRS/IPSN and ENUSA methods. In the case of UMAE, no uncertain input parameters must be specified: whatever is the selected list of parameters (provided their number is sufficiently large, ref. /24/), the uncertain results will not change.

All methodologies require, to different extents, that scaling analysis be made, item 6). This is done at qualitative level during code validation and may require specific analyses during the application of the methodology. In the case of UMAE the demonstration that a given physical phenomenon occurs in differently scaled facilities is necessary: this can be achieved through the use of the FFT (Fast Fourier Transform) based method, ref. /29/. Owing to the above, the UMAE could not be applied if experiments reproducing the target test scenario are not available.

Experts are needed by all methodologies, e.g. for the optimal use of the code, for ranking the importance of phenomena or for selecting relevant experimental data. However, expert judgements are not needed by users of UMAE, or if used, their influence is controlled by fixed targets of accuracy (this means that unexperienced code users, will never come out from the loop included in the procedure that imposes upper limit for accuracy; Expert Panels are never necessary in the methodology).
Standardised expert judgements are included in the method as defined. Biases on output values should be avoided or reduced to the minimum extent to prevent the adding of subjective judgements on the output uncertainty, item 9). CSAU considers the biases on output value a cost efficient way to reduce the code runs. In the case of UMAE, calculation of biases may come from the lack of consideration of phenomena expected in the reference transient and not present in the data base: an example is the nuclear fuel performance when all the considered data base comes from facilities equipped with electrically heated fuel rod simulators.

In relation to Tab. 4:
Prioritization of input parameters is supported by a ranking or brainstorming processes in the case of AEAT and CSAU, item 3). The state of knowledge of input uncertain parameters is expressed, where applicable, by a subjective probability distribution, item 4). Expert judgement is needed in the case of GRS and CSAU to account for the state of knowledge. In the case of UMAE, the knowledge of parameter uncertainty is not considered; however, significant error may cause failure of the process. Statistical rigorous algorithms are part of GRS, IPSN and ENUSA methodologies, item 6). This is not the case of UMAE; however, the influence of statistics related algorithms upon the results is much reduced in the UMAE compared with the other methods. The statistics based methods, GRS, IPSN and ENUSA allow the possibility to evaluate the influence of a reduced number of calculations, item 11), and of the most important input uncertain parameters on the output uncertainty, item 12). This is not possible in the case of UMAE. A similar situation occurs in relation to the characterization of the relevance of input uncertain parameters, as far as output uncertainty is concerned, item 13).

5. CONCLUSIONS

The uncertainty ranges of key output parameters from system thermalhydraulic code calculations reflect the state of knowledge of plant transient prediction; the calculation of such ranges is mandatory for best estimate codes. The need of uncertainty analysis in system code calculations stimulated the present study. Several sources of uncertainty have been classified (first goal of the paper); part of these are embedded into the currently adopted codes and part come out in the frame of their applications. Any calculation from a best estimate code, to be meaningful, needs an uncertainty evaluation: this is valid for various applications of the codes ranging from licensing studies to training of plant operators.

Five methodologies have been considered that allow the quantification of uncertainty in code results when predicting Nuclear Plant scenarios: these recently underwent a thorough assessment in the frame of the UMS (Uncertainty Methods Study) promoted by the OECD/CSNI. The five methodologies are identified as AEAT, GRS, IPSN, ENUSA and UMAE. The characterization of the uncertainty methodologies constitutes the second goal of the paper; the following summary observations apply:
AEAT: Each step of the methodology is under strict control of experts that
do not use statistics or probability distributions for input uncertain parameters. As a consequence, the most relevant uncertainty sources must be identified to avoid an impractical number of code runs.

GRS: The full use of statistics makes the computational effort necessary to obtain uncertainty, independent upon the number of selected input parameters. Subjective probability density functions must be supplied in relation to each uncertain input parameter.

IPSN: This is basically similar to the previous one. An effort must be made to exclude or minimize the nodalization effects on the uncertainty.

ENUSA: This is derived from the CSAU; so it needs a screening of relevant phenomena in a way similar to AEAT; the PIRT process is applied here. The processing of the identified input uncertainty parameters is performed in a way similar to what done by GRS and IPSN.

UMAE: In this case, as a difference from the previous methodologies, the output of relevant code results are processed and uncertainty is derived by the propagation of accuracy. Subjective judgements are almost avoided in the method application, but relevant experimental data must be available.

REFERENCES


/8/ Bucalossi A., D'Auria F.: "Fundamental assessment of system codes: Cathare 2 v.l.3E" - University of Pisa Report, DCMN, NT 277(96), Pisa (I), May 1996

/9/ Trambauer K.

450
/13/ D'Auria F., Galassi G.M., Madeira A.: "Effect of minor boundary conditions on dryout during small LOCA in PWRs" - X Brazilian Meet. on Reactor Physics and Thermalhydraulics, Agua de Lindoia, SP, Aug. 7-11, 1995
/19/ Holmstrom H.: "Quantification of Code Uncertainties" - CSNI Specialists Meeting on Transient Thermalhydraulics - Aix-en-Provence (F), April 6-9, 1992


Tab. 1 - Influence of user effect at the level of input deck (nodalization) development. Considered case is the Lobi natural circulation, ref. /1/.

- RESULTS AT 240s INTO THE TRANSIENT -

<table>
<thead>
<tr>
<th>SUCCESSFUL ADV.</th>
<th>IBM 3090 F CROCE - PISA</th>
<th>IBM 3090 ENDA - BO</th>
<th>ANDAL JRC - INFRA</th>
<th>CRAY JUD JEND - PISA</th>
<th>FACOM JAXRI</th>
<th>PC UNIV. ZAGREB</th>
</tr>
</thead>
<tbody>
<tr>
<td>CPU TIME</td>
<td>13393</td>
<td>13094</td>
<td>13565</td>
<td>13804</td>
<td>14899</td>
<td>13343</td>
</tr>
<tr>
<td>CPU TIME</td>
<td>5227</td>
<td>5940</td>
<td>4815</td>
<td>2919</td>
<td>635.2</td>
<td>29330.7</td>
</tr>
<tr>
<td>MASS ERROR PS</td>
<td>-0.882</td>
<td>-0.802</td>
<td>-0.800</td>
<td>-1.259</td>
<td>-0.904</td>
<td>-0.077</td>
</tr>
<tr>
<td>MASS ERROR BS</td>
<td>0.153</td>
<td>0.150</td>
<td>0.153</td>
<td>0.196</td>
<td>0.166</td>
<td>0.155</td>
</tr>
<tr>
<td>PRE PRESSURE</td>
<td>7.397</td>
<td>7.386</td>
<td>7.405</td>
<td>7.717</td>
<td>7.638</td>
<td>7.389</td>
</tr>
<tr>
<td>SG SO PRESSURE</td>
<td>7.030</td>
<td>7.030</td>
<td>7.832</td>
<td>7.866</td>
<td>7.863</td>
<td>7.833</td>
</tr>
<tr>
<td>PS MASS</td>
<td>9969.4</td>
<td>9970.0</td>
<td>9970.8</td>
<td>9966.3</td>
<td>9929.1</td>
<td>9968.4</td>
</tr>
<tr>
<td>ES MASS</td>
<td>4613.1</td>
<td>4609.5</td>
<td>4613.4</td>
<td>4618.0</td>
<td>4670.1</td>
<td>4614.4</td>
</tr>
<tr>
<td>CORE DP</td>
<td>14.3</td>
<td>14.5</td>
<td>14.5</td>
<td>15.3</td>
<td>12.2</td>
<td>14.1</td>
</tr>
<tr>
<td>VOID GC CHECK (1)</td>
<td>0.211</td>
<td>0.221</td>
<td>0.212</td>
<td>0.158</td>
<td>0.118</td>
<td>0.217</td>
</tr>
<tr>
<td>VOID GC CHECK (6)</td>
<td>0.656</td>
<td>0.660</td>
<td>0.660</td>
<td>0.604</td>
<td>0.619</td>
<td>0.664</td>
</tr>
<tr>
<td>VOID GC CHECK (10)</td>
<td>0.650</td>
<td>0.653</td>
<td>0.655</td>
<td>0.627</td>
<td>0.999</td>
<td>0.645</td>
</tr>
</tbody>
</table>

Tab. 2 - Influence of computer/compiler effect on code calculation results: considered case is the OECD/CSNI International Standard Problem 26, ref. /10/.
<table>
<thead>
<tr>
<th>No</th>
<th>General Characteristic</th>
<th>AEAT</th>
<th>CSAU/ENUSA</th>
<th>GRS/IPSN</th>
<th>UMCE</th>
</tr>
</thead>
<tbody>
<tr>
<td>1</td>
<td>Restriction on the number of input uncertain parameters</td>
<td>Y</td>
<td>Y</td>
<td>N(1)</td>
<td>na</td>
</tr>
<tr>
<td>2</td>
<td>Deriving input uncertainty ranges</td>
<td>Y</td>
<td>Y</td>
<td>Y</td>
<td>N</td>
</tr>
<tr>
<td>3</td>
<td>Assigning subjective probability distributions</td>
<td>N</td>
<td>Y</td>
<td>Y</td>
<td>N</td>
</tr>
<tr>
<td>4</td>
<td>Use of statistics</td>
<td>N</td>
<td>Y</td>
<td>Y</td>
<td>Y(2)</td>
</tr>
<tr>
<td>5</td>
<td>Use of response surface technique</td>
<td>N</td>
<td>Y/N</td>
<td>N</td>
<td>N</td>
</tr>
<tr>
<td>6</td>
<td>Need of specific data for scaling</td>
<td>N</td>
<td>Y(3)/N</td>
<td>N</td>
<td>Y</td>
</tr>
<tr>
<td>7</td>
<td>Quantification of code calculation accuracy</td>
<td>N</td>
<td>N</td>
<td>N</td>
<td>Y</td>
</tr>
<tr>
<td>8</td>
<td>Use of expert groups</td>
<td>Y</td>
<td>Y</td>
<td>Y(2)</td>
<td>N</td>
</tr>
<tr>
<td>9</td>
<td>Need of bias on the results</td>
<td>N</td>
<td>Y/N</td>
<td>N</td>
<td>Y</td>
</tr>
</tbody>
</table>

Y = Yes; N = No; na = not applicable

(1) In the case of GRS, the possibility to modify some discrete parameters is accepted.
(2) To a limited extent.
(3) At a qualitative level during code validation.
(4) All methodologies require data for scaling during code validation.

Tab. 3 - Main characteristics of the considered uncertainty methods.

<table>
<thead>
<tr>
<th>No</th>
<th>Additional Feature</th>
<th>AEAT</th>
<th>CSAU/ENUSA</th>
<th>GRS/IPSN</th>
<th>UMCE</th>
</tr>
</thead>
<tbody>
<tr>
<td>1</td>
<td>Determination of input uncertain parameters and related ranges</td>
<td>EX</td>
<td>EX</td>
<td>EX(1)</td>
<td>na(2)</td>
</tr>
<tr>
<td>2</td>
<td>Selection of uncertain parameter values within the determined ranges</td>
<td>EX</td>
<td>EX/RS</td>
<td>RS</td>
<td>na</td>
</tr>
<tr>
<td>3</td>
<td>Support to identification and ranking of input uncertain parameters</td>
<td>Y</td>
<td>Y</td>
<td>N</td>
<td>N</td>
</tr>
<tr>
<td>4</td>
<td>Account for state of knowledge of input uncertain parameters</td>
<td>N</td>
<td>Y-EX</td>
<td>Y-EX</td>
<td>N</td>
</tr>
<tr>
<td>5</td>
<td>Probabilistic uncertainty statement</td>
<td>N</td>
<td>Y</td>
<td>Y</td>
<td>Y</td>
</tr>
<tr>
<td>6</td>
<td>Statistical rigour</td>
<td>na</td>
<td>Y</td>
<td>Y</td>
<td>N</td>
</tr>
<tr>
<td>7</td>
<td>Knowledge of code features reduces resources necessary for the analysis</td>
<td>Y</td>
<td>Y</td>
<td>N</td>
<td>N</td>
</tr>
<tr>
<td>8</td>
<td>Number of code runs independent from number of input and output parameters</td>
<td>N</td>
<td>N/Y</td>
<td>Y</td>
<td>na</td>
</tr>
<tr>
<td>9</td>
<td>Typical number of needed code runs</td>
<td>N</td>
<td>N/Y</td>
<td>Y</td>
<td>N</td>
</tr>
<tr>
<td>10</td>
<td>Typical number of uncertain input parameters</td>
<td>10</td>
<td>10(5)</td>
<td>50</td>
<td>na</td>
</tr>
<tr>
<td>11</td>
<td>Quantitative information about the influence of a limited number of runs</td>
<td>N</td>
<td>N/Y</td>
<td>Y</td>
<td>N</td>
</tr>
<tr>
<td>12</td>
<td>Sensitivity measures of input parameters on output parameters</td>
<td>N</td>
<td>N/Y</td>
<td>Y</td>
<td>N</td>
</tr>
</tbody>
</table>

Y = Yes; N = No; EX = Expert judgement needed; RS = Random Selection according to subjective probability distribution; na = not applicable

(1) In the case of GRS, the number and the type of input parameter is unlimited.
(2) Differences between relevant measured and calculated trends determine the ranges.
(3) Only one if one considers the NPP nodalization; however, around 10 if ITF calculations are considered.
(4) Order of magnitude.
(5) Very much depending upon the application.

Tab. 4 - Additional features of the considered uncertainty methods.
Fig. 1 - Comparison between results of the original Shah correlation for condensation heat transfer coefficient and Shah correlation after implementation in the Cathare code (two different steam velocities), ref. /8/.

Fig. 2 - Code user effect when developing a nodalization or interpreting data from experimentalists, ref. /1/.

Fig. 3 - Code user effect when accepting a steady state, ref. /1/.
Fig. 4 - Overall code user effect on a system calculation: considered case is the OECD/CSNI International Standard Problem 26, ref. /1/.

a)

Fig. 5 - Influence of a nodalization inadequacy on code results. Considered case is the OECD/CSNI International Standard Problem 18, ref. /12/.
Fig. 6 - Influence of boundary conditions (heat losses from SG) on relevant code results (rod surface temperature and time of accumulator actuation); considered case is a small break LOCA experiment in LOBI facility, refs. /10/ and /13/.
Fig. 7 - Loss of feedwater transient scenario in Lobi facility in relation to which a code model deficiency has been identified, ref. /14/.

Fig. 8 - Loss of feedwater transient scenario in Lobi facility: comparison between predicted trends and measured trend of SG downcomer level, allowing the characterization of the code deficiency, ref. /14/.
Application of Uncertainty Methods in the OECD/CSNI
Uncertainty Methods Study

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4) Universita Degli Studi di Pisa, Italy
5) Empresa Nacional del Uranio, SA, Spain

OECD/CSNI Seminar on
„Best Estimate Methods in Thermal-Hydraulic Safety Analysis“
Ankara, Turkey, 29 June - 1 July 1998
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Abstract

The Uncertainty Methods Study (UMS) Group, following a mandate from CSNI, has compared five methods for calculating the uncertainty in the predictions of advanced "best estimate" thermal-hydraulic codes.

One of the methods, the Pisa method, is based on extrapolation from-integral experiments. To use this method stringent criteria on the code and experimental data base must be met. The other methods identify and combine input uncertainties. Three of these, the GRS, IPSN and ENUSA methods, use subjective probability distributions and one, the AEAT method, performs a bounding analysis.

Each method has been used to calculate the uncertainty in specified parameters for the LSTF SB-CL-18 5% cold leg small break LOCA experiment in the Large Scale Test Facility (LSTF). The uncertainty analysis was conducted essentially blind and the participants did not use experimental measurements from the test as input apart from initial and boundary conditions. Participants calculated uncertainty ranges for experimental parameters including pressuriser pressure, primary circuit inventory and clad temperature (at a specified position) as functions of time.

461
Following the objectives of the study, the participants have:

- compared the methods, step by step, when applied to LSTF SB-CL-18
- compared the uncertainties predicted with each other, and
- compared the uncertainties predicted with the measured values.

Where the predictions of the methods differ, the differences have been accounted for in terms of the assumptions of the methods and the input data used. The calculated ranges bound the experimental results with some exceptions. The possible causes of these discrepancies have been identified.

Based on this study of one experiment, the UMS group has concluded that the methods are suitable for use. The major differences between the predictions of the methods came from the quantification of the input uncertainties and consequently the wideness of the uncertainty ranges and the choice of uncertain parameters. For the Pisa method differences come mainly from the optimisation of the nodalisation and possibly from the different number of experiments investigated. In all cases appropriate technical knowledge, skill and expertise should be exercised when running the "best estimate" codes and performing uncertainty analysis and the appropriate quality standards must be applied. Uncertainty analysis is needed if useful conclusions are to be obtained from "best estimate" thermal-hydraulic codes.
1 Introduction

Some computer codes that model reactor accidents contain deliberate pessimisms and unphysical assumptions. It is then argued that the overall predictions are worse than reality (for example, calculated fuel temperatures are higher than reality). These conservative Evaluation Models, for example, were provided to satisfy US licensing requirements for such pessimistic calculations up to 1988/1/.

The other approach is to design a code to model all the relevant processes in a physically realistic way with the intention that the predictions are not biased either in a pessimistic or optimistic direction ("best estimate"). The main motivations for the development of advanced best estimate thermal hydraulic codes were:

- To describe the physical behaviour realistically without individual conservative assumptions which may not necessarily result in conservative results.

- To obtain more economical operation of reactors by relaxing technical specifications and core operating limits set by conservative Evaluation Model calculations. This might include prolonging fuel cycles, up-rating power and justifying the continued operation of some reactors against modern safety standards.

- To develop effective accident management measures based on realistic analysis.

These realistic computer codes can approximate the physical behaviour with more or less accuracy. The inaccuracies are stated during the usual code validation process. Agreement of calculated results with experimental data is often obtained by choosing specific code input options or by changing parameters in the model equations. These selected parameters usually have to be changed again for different experiments in the same facility or a similar experiment in a different facility in order to obtain agreement with data. A single unbiased calculation with an advanced realistic code would give results of one code run of unknown accuracy. To make such results useful, for example if they are to be compared with limits of acceptance, the uncertainty in the predictions then has to be calculated separately. Uncertainty analysis methods therefore had to be developed to estimate safety to justify reactor operation and to help prepare Emergency Response Guidelines for accident management /2/. A particular stimulus was the USNRC's revision of its regulations in 1989 to permit the use of realistic models with quantification of uncertainty in licensing submissions for Emergency Core Cooling Systems /3/.

In addition uncertainty analysis can be used to assist research prioritisation. It can help to identify models that need improving and areas where more data are needed. It can make code development and validation more cost-effective.

The Uncertainty Methods Study (UMS) compares different methods to estimate the uncertainty in predictions of advanced best estimate thermal hydraulic codes by applying the methods to a particular experiment. This paper summarises the comparison reported in reference /4/.
2 Objectives of the Uncertainty Methods Study

The CSNI Task Group on Thermal Hydraulic System Behaviour held a Workshop to discuss the different uncertainty methods in 1994. The Workshop recommended that there should be an international comparison exercise between the available uncertainty analysis methods with the objectives /2/: 

1. To gain insights into differences between features of the methods by:
   - comparing the different methods, step by step, when applied to the same problem
   - comparing the uncertainties predicted for specified output quantities of interest
   - comparing the uncertainties predicted with measured values
   and so allowing conclusions to be drawn about the suitability of methods.

2. To inform those who will take decisions on conducting uncertainty analyses, for example in the light of licensing requirements.

The CSNI approved the Uncertainty Methods Study (UMS) at its meeting in December 1994, with these objectives. The UK was given the task of leading the study.

3 Methods Compared

The methods compared in the UMS are summarised in Table 1. The methods (and those discussed at the 1994 Workshop) may be divided into three groups according to their basic principles /4, 5/: 

- The University of Pisa method, the Uncertainty Method based on Accuracy Extrapolation (UMAE), extrapolates the accuracy of predictions from a set of integral experiments to the reactor case or experiment being assessed.

The other methods rely on identifying uncertain models and data and quantifying and combining the uncertainties in them. They fall into two kinds:

- The AEA Technology method, that characterises the uncertainties by "reasonable uncertainty ranges" and attempts to combine these ranges with a bounding analysis and

- Methods which assign subjective probability distributions to uncertainty ranges for uncertain input parameters and sample the resulting subjective probability density at random in the space defined by the uncertainty ranges. In the UMS, the GRS, IPSN and ENUSA methods are of this kind; so also is the Code Scaling Applicability and Uncertainty Evaluation (CSAU) Method as demonstrated in /6/. The term "subjective" is used here to distinguish between subjective probability due to imprecise knowledge and probability due to stochastic or random variability.
Table 1: Summary of Methods Compared in the UMS Study

<table>
<thead>
<tr>
<th>Participant</th>
<th>Code Version Used</th>
<th>Method Name and Type</th>
</tr>
</thead>
<tbody>
<tr>
<td>AEA Technology, UK</td>
<td>RELAP5/MOD3.2</td>
<td>AEAT Method. Phenomena uncertainties selected, quantified by ranges and combined.</td>
</tr>
<tr>
<td>University of Pisa</td>
<td>RELAP5/MOD2 cycle 36.04, IBM version</td>
<td>Uncertainty Method based on Accuracy Extrapolation (UMAE). Accuracy in calculating similar integral tests is extrapolated to plant.</td>
</tr>
<tr>
<td>Gesellschaft für Anlagen- und Reaktorsicherheit (GRS), Germany</td>
<td>ATHLET Mod 1.1 Cycle A</td>
<td>GRS Method. Phenomena uncertainties quantified by ranges and subjective probability distributions (SPDs) and combined.</td>
</tr>
<tr>
<td>Institut de Protection et de Sûreté Nucleaire (IPSN), France</td>
<td>CATHARE 2 version 1.3U rev 5</td>
<td>IPSN Method. Phenomena uncertainties quantified by ranges and SPDs and combined.</td>
</tr>
<tr>
<td>Empresa Nacional del Uranio, SA (ENUSA), Spain</td>
<td>RELAP5/MOD 3.2</td>
<td>ENUSA Method. Phenomena uncertainties quantified by ranges and SPDs and combined.</td>
</tr>
</tbody>
</table>

4 Chosen Experiment

The participants and the Task Group of Thermal Hydraulic System Behaviour agreed that the International Standard Problem (ISP) 26 experiment, LSTF SB-CL-18, should be used. LSTF SB-CL-18 is a 5% cold leg small break LOCA experiment conducted in the ROSA-IV Large Scale Test Facility (LSTF). The LSTF is located at the Tokai Research Establishment of JAERI and is a 1/48 volumetrically scaled, full height, full pressure simulator of a Westinghouse type 3423 MW_e PWR. The experiment simulates a loss of off-site power, no high pressure injection (HPIS), the accumulator system initiates coolant injection into the cold legs at a pressure of 4.51 MPa, the low pressure injection system initiates at 1.29 MPa. Thus, the experiment considers a beyond design basis accident. Because of the ISP the experiment was already well documented. Much helpful information has been provided by Yutaka Kukita of the Japan Atomic Energy Research Institute (JAERI).
Although ISP 26 was an open ISP participants in the UMS have not used experimental measurements from the test, for example the break flow or the secondary pressure. In the same way other experiments performed in the LSTF were not used. This means not allowing the related measurements to influence the choice of input uncertainty ranges and subjective probability distributions of the input uncertainties. Submissions included a written statement of the justification used for input uncertainty ranges and distributions used and were reviewed at Workshops. All other assumptions made were listed and reviewed.

5 Step-by-step Comparison of the Methods

A detailed description of the methods is given in references /4, 5/. Table 2 sets out a step by step comparison and displays the main similarities and differences between the methods.

6 Comparison of Calculated Uncertainty Ranges

The participants were asked to make uncertainty statements, based on their calculation results, for:

1 Functions of time
   - Pressuriser pressure
   - Primary circuit mass inventory
   - Rod surface temperature for B18 rod (4, 4) position 8 (TW359 TWE-B18448).

2 Point quantities
   - First peak clad temperature
   - Second peak clad temperature
   - Time of overall peak clad temperature (the higher of the two)
   - Minimum core pressure difference at DP 50 DPE300-PV.

As a result of the short time-scale of UMS and limited funding some of the uncertainty ranges derived from experimental data for the calculations have been less well refined and, as a consequence, wider than would have been the case, for example, if the calculations were for a plant safety analysis. This affected the AEAT and ENUSA counter-current flow (CCFL) ranges, for example.
<table>
<thead>
<tr>
<th>Feature</th>
<th>AEA Technology</th>
<th>University of Pisa</th>
<th>Probabilistic methods - GRS, IPSN, ENUSA</th>
</tr>
</thead>
<tbody>
<tr>
<td>Characterisation of uncertainties</td>
<td>Reasonable uncertainty ranges</td>
<td>Accuracy: difference between prediction and measurement</td>
<td>Ranges and subjective probability distributions</td>
</tr>
<tr>
<td>Selection of important uncertainties</td>
<td>Yes (9 and 7 in two demonstration studies)</td>
<td>No</td>
<td>GRS, IPSN: No (52 and 25 here) ENUSA: PIRT (25)</td>
</tr>
</tbody>
</table>
| Combination and propagation of uncertainties | Analyst explores uncertainty space and decides when to stop | Measured accuracy can be extrapolated to plant if:  
  - criteria on integral test data satisfied  
  - data, nodalization, users are qualified  
  - relevant thermal hydraulic aspects are in data base  
  - measured/calculated ratio scattered about 1.0; | Uncertainty space sampled at random according to combined subjective probability distribution |
<table>
<thead>
<tr>
<th>Feature</th>
<th>AEA Technology</th>
<th>University of Pisa</th>
<th>Probabilistic methods - GRS, IPSN, ENUSA</th>
</tr>
</thead>
<tbody>
<tr>
<td>Number of code runs</td>
<td>22 and 50 in two studies</td>
<td>One for each test in the data base and one for the plant</td>
<td>For one sided 95%/95% tolerance/confidence interval: 59; for two-sided interval: 93</td>
</tr>
<tr>
<td>Use of specific data for scaling</td>
<td>During code validation</td>
<td>Yes</td>
<td>During code validation</td>
</tr>
<tr>
<td>Use of response surface to approximate result</td>
<td>No</td>
<td>No</td>
<td>No</td>
</tr>
<tr>
<td>Use of biases on results</td>
<td>No</td>
<td>Possibly</td>
<td>No</td>
</tr>
<tr>
<td>Continuous output parameters</td>
<td>Overall maxima and minima of calculated curves or judged envelope curves if wider</td>
<td>By linear interpolation</td>
<td>Overall maxima and minima of calculated curves</td>
</tr>
<tr>
<td>Sensitivity of output parameters to input parameters</td>
<td>Understanding of the processes and their interactions generated</td>
<td>No</td>
<td>Statistical sensitivity information generated</td>
</tr>
<tr>
<td>Check against independent experimental database</td>
<td>Yes</td>
<td>Yes</td>
<td>GRS: yes IPSN, ENUSA: optional</td>
</tr>
</tbody>
</table>
6.1 Functions of Time

The uncertainty ranges calculated for pressuriser pressure, primary inventory and hot rod surface temperature for LSTF B18 rod (4,4) at position 8 are shown in Figures 1, 2 and 3.

6.1.1 Clad Temperature Ranges

The AEAT and ENUSA predictions are remarkably similar. The same code (RELAP5), the same base input deck and mostly the same input uncertainties were used. However, they applied different methods, i.e. bounding (AEAT) versus probabilistic treatment (ENUSA) of the uncertainties, and the numbers of the input uncertainties are different (7 by AEAT and 25 by ENUSA).

The two applications of the Pisa (UMAE) method using different computer codes (RELAP and CATHARE) are broadly similar. The clad temperature ranges for the second dry-out are similar. Those ranges for the first dry-out calculated by CATHARE gave a narrower range because the base case calculation did not predict the dry-out. However, the UMAE still generated a successful uncertainty range for this quantity because of the first dry-out measured on the experiments considered for accuracy extrapolation to the LSTF experiment.

Of the probabilistic methods, IPSN gave the narrowest ranges. This is attributed to the choice of parameters made and to the parameter ranges used. An actuation of the CCFL option in the CATHARE code (at the upper core plate and at the inlet plena of the steam generators) was needed to predict the first dry-out.

The large uncertainty ranges for clad temperature calculated by AEAT, ENUSA and GRS prompted discussion among the participants about the reasons for them.

In the case of AEAT and ENUSA this may be in part due to unrealistically large uncertainty ranges for CCFL parameters, agreed upon in order to complete the study on time. When AEAT discounted CCFL uncertainty completely the calculated uncertainty range for peak clad temperature changed from 584 - 1142 K to 584 - 967 K.

In the experiment the measured maximum clad temperature during the second heat-up is well below that of the first. In the GRS calculations, many of the 99 calculated time histories exhibit the contrary, namely a maximum during the second heat-up. The uncertainty of the maximum during the second heat-up is about double of that during the first. In some code runs a partial dry-out occurs due to fluid stagnation in the rod bundle which causes an earlier second heat-up and as a consequence a larger increase of the clad temperature.

The large uncertainty is striking, and the sensitivity measures indicate which of the uncertainties are predominantly responsible (see Volume 2 of reference /4/). Main contributions to uncertainty come from the critical discharge model and the drift in the heater rod bundle. Other contributions that are worth mentioning come from the by-
pass cross section upper downcomer - upper plenum. As a consequence, improved state of knowledge about the respective parameters would reduce the striking uncertainty of the computed rod surface temperature in the range of the second core heat-up most effectively.

### 6.1.2 Missing First Dry-outs

AEAT did not calculate the first dry-out (observed at 150 s) in many runs, particularly near the top of the core. AEAT attribute this to not having treated uncertainty in critical heat flux (CHF). They did not consider uncertainty in CHF modelling because they expected the uncertainty in overall peak clad temperature to be dominated by other uncertainties (which was, in fact, the case).

All the University of Pisa CATHARE calculations were performed with the frozen version of the code released by the developers with standard/default code options. The related treatment of CCFL was partly responsible for the misprediction together with other reasons, such as the distribution of pressure drops along the entire primary loop, which have not been fully investigated, considering that the quantitative judgement of the results was better than the fixed acceptability threshold for proceeding with the uncertainty analysis.

The IPSN CATHARE reference calculation, using the CCFL model at the upper core plate, calculated only a very slight heat up (\( \sim 10^\circ C \)) at the top of the core (elevation 8). The uncertainties relative to several parameters (CCFL parameters, heat transfer coefficients . . . . . .) led to about 150°C of uncertainty range for this heat-up.

### 6.1.3 Late Quench Times and Third Dry-outs

Participants questioned the realism of late quench times, up to about 600 s. AEAT analysed cases calculated where quench times at level 8 were after 570 s. This occurred when:

- there was a third dry-out or
- loop seal B (broken loop) does not clear (or clears late) which is associated with low break flow or modelling the off-take at the top of the cold leg or
- in one case where loop seal A (intact loop) did not clear and CCFL parameters that maximise the hold-up of water (lower bound of c and upper bound of \( m \) in the Wallis correlation) were used in the steam generator (SG) and SG inlet plenum or
- in other cases where the CCFL parameters that maximise the hold-up of water were used in the SG and SG inlet plenum and break flow was mostly relatively low.

In 17 out of the 50 cases reported by AEAT a third dry-out was detected. The third dryouts lead to relatively small temperature transients which are localised at the top of the core. They are not correlated with high peak clad temperatures. AEAT therefore further examined the late quench cases without third dry-outs. Some of these had maximum CCFL or break take-off modelled at the top of the pipe. Further work may refine these input uncertainty ranges. However, other late quench cases did not de-
pend on such inputs. These cases had nominal CCFL parameters and low break flow and late or no clearance of the loop seal in the broken loop (loop B).

Third dry-outs were also calculated by ENUSA and GRS. GRS found a third dry-out in 2 out of 99 calculations. They were associated with high bypass flow cross section and high condensation. High condensation keeps the pressure low so the accumulators inject continuously. However the accumulator water is all discharged through the break. Although the input uncertainties leading to these cases differ from those found by AEAT they also do not calculate clearance of the broken loop seal. Some other GRS calculations without loop seal clearance do not predict a third dryout, however.

6.2 Point quantities

Table 3 compares the uncertainty ranges for the selected point quantities.

Table 3: Uncertainty Ranges for Point Quantities

<table>
<thead>
<tr>
<th>Quantity</th>
<th>Limit</th>
<th>AEAT</th>
<th>RELAP5</th>
<th>CATHARE</th>
<th>Pisa</th>
<th>GRS</th>
<th>IPSN</th>
<th>ENUSA</th>
<th>Measured Value</th>
</tr>
</thead>
<tbody>
<tr>
<td>PCT1 (K)</td>
<td>upper</td>
<td>938</td>
<td>773</td>
<td>693</td>
<td>904</td>
<td>813</td>
<td>1082</td>
<td>1075</td>
<td>740</td>
</tr>
<tr>
<td></td>
<td>lower</td>
<td>573</td>
<td>559</td>
<td>563</td>
<td>580</td>
<td>693</td>
<td>555</td>
<td></td>
<td></td>
</tr>
<tr>
<td>PCT2 (K)</td>
<td>upper</td>
<td>1142</td>
<td>620</td>
<td>681</td>
<td>849</td>
<td>653</td>
<td>1101</td>
<td>1099</td>
<td>610</td>
</tr>
<tr>
<td></td>
<td>lower</td>
<td>584</td>
<td>519</td>
<td>511</td>
<td>560</td>
<td>503</td>
<td>609</td>
<td></td>
<td></td>
</tr>
<tr>
<td>t_{PCT1}</td>
<td>upper</td>
<td>332</td>
<td>232</td>
<td>195</td>
<td>192</td>
<td>165</td>
<td>302</td>
<td>300</td>
<td>150</td>
</tr>
<tr>
<td></td>
<td>lower</td>
<td>168</td>
<td>124</td>
<td>135</td>
<td>84</td>
<td>120</td>
<td>180</td>
<td></td>
<td></td>
</tr>
<tr>
<td>t_{PCT2}</td>
<td>upper</td>
<td>516</td>
<td>548</td>
<td>630</td>
<td>564</td>
<td>525</td>
<td>642</td>
<td>641</td>
<td>500</td>
</tr>
<tr>
<td></td>
<td>lower</td>
<td>280</td>
<td>396</td>
<td>322</td>
<td>400</td>
<td>431</td>
<td>254</td>
<td></td>
<td></td>
</tr>
<tr>
<td>ΔP_{min}</td>
<td>upper</td>
<td>5.55</td>
<td>15.3</td>
<td>2.2</td>
<td>12.0</td>
<td>3.41</td>
<td>3.8</td>
<td></td>
<td></td>
</tr>
<tr>
<td></td>
<td>lower</td>
<td>0.63</td>
<td>16.0</td>
<td>0.75</td>
<td>0.44</td>
<td></td>
<td></td>
<td></td>
<td></td>
</tr>
</tbody>
</table>

(1) converted from collapsed liquid level
(2) for level 8 only. The other PCT values are maximised over all levels.

The results in this table may be compared with the trends discussed previously for functions of time. For peak clad temperatures the same trends are observed except that:

- Out of the probabilistic methods there is a larger variation between the first PCT ranges. This is probably because the peaks are at different times.

- The different predictions for the range of time of PCT is strongly affected by whether each method predicts the overall PCT to be at the first peak or the second one or either.

- The lower ends of the uncertainty ranges for ΔP_{min} are related to core dryout. However the correlation with PCT is slight because PCT is also strongly affected by the extent in time of core uncovering. The upper ends of the ΔP_{min} ranges come from combinations of the more benign values of the input uncertainties.
7 Conclusions and Recommendations

Five uncertainty methods for advanced best estimate thermal hydraulic codes have been compared. The objectives of this international UMS have been fully accomplished, in particular:

- A step by step comparison of the five different methods has been carried out.

- The predicted uncertainties have been compared with each other and with the measured results.

- The uncertainties predicted by each method have been compared. Where the predictions differ, the differences have been accounted for in terms of the assumptions of the methods and the input data used.

- The results of each method cover the experimental results with some minor exceptions, for which the possible sources of discrepancies have been identified.

- The five methods have proved to be suitable for quantifying uncertainties with best estimate codes, in this particular application.

7.1 Comparison between the Calculations

The differences between the predictions from the methods will come from a combination of:

- the method used
- the underlying accuracy of the modelling used
- the completeness of the identification and selection of uncertainties
- the conservatism of the input (e.g. uncertainty ranges, subjective probability distributions) used and
- in the Pisa method optimisation of the nodalisation and possibly from the different number of experiments investigated (5 experiments with CATHARE, 10 experiments with RELAP).

Of these, differing degrees of conservatism of the input, particularly where data are sparse, probably account for the major differences between the specified uncertainty ranges. For example, critical two-phase flow data from LSTF were not used, and data for the geometry of the LSTF orifice were not available.

The ideal is for a specified uncertainty range that fully reflects the accuracy of the underlying modelling. The input uncertainties mainly responsible for the differences are probably those identified as important by the sensitivity studies of the methods. These are: Choked flow and counter-current flow limitation (AEAT and ENUSA), and choked
flow and drift in the heater rod bundle (GRS). Further work to refine input uncertainty ranges for these is most likely to affect the calculated uncertainty ranges.

It follows that, it is very important how the methods are applied.

7.2 Guidelines on Choice of Methods

The choice of an uncertainty method depends on what the end users will accept and in what features they are interested.

The Pisa Method can be used if:
- stringent criteria on data and modelling accuracy are met;
- end users accept assumptions about extrapolation from small scale experimental facilities to large scale;
- most likely to be the case when extrapolation to the case of interest is small.

Probabilistic methods (GRS, IPSN, ENUSA) can be used if:
- end users accept the method based on combination of subjective probability distributions (SPDs);
- users are interested in sensitivity measures to help prioritise future improvements;
- users are not interested to limit the number of input uncertainties because the number of code calculations to be performed is independent of the number of uncertainties.

The AEAT method can be used if:
- end users prefer not to rely on assigned SPDFs;
- users rely on skill of analysts to find maxima and minima;
- method yields understanding of the interactions between processes to help prioritise future improvements.

In all cases appropriate knowledge, skill, experience and quality standards must be applied.

7.3 Other Conclusions

Uncertainty analysis is needed if useful conclusions are to be obtained from "best estimate" thermal-hydraulic code calculations, otherwise point values of unknown accuracy would be presented for comparison with limits for acceptance. Code facilities needed for uncertainty analysis include, for example, the provision for varying models and parameters.
For some of the methods described here the present study was the first full application. For all of them it constituted a pilot study that allowed a better understanding of the assumptions at the basis of the method.

The information gained in the UMS about the methods described and how they can be applied, can be used to inform decisions on the conduct of uncertainty analyses, for example in the light of licensing requirements.

8 Acknowledgements

The work of AEA Technology was supported by the UK Health and Safety Executive.

A number of scientists of the University of Pisa and of foreign Organisations contributed to the development of the UMAE. Co-operation with the University of Zagreb, mainly during 1989-94, allowed the finalization of a large number of activities in the frame of the UMAE (Dr. N. Debrick and Dr. T. Bajs had the main role). The completion of PhD theses by M. Leonardi, S. Belzito and M. Ingegneri contributed to the progress of UMAE activities, as did the PhD theses of E. Eramo, P. Gatta and W. Giannotti still in progress. Recent international co-operations in the UMAE framework with CNEN of Rio de Janeiro (Dr R. Galetti and Ms A. Madeira), NPIC of Chengdu (Dr H. Zhao), IJS of Ljubljana (Prof. B. Mavko and Mr A. Prosek), IPSN of Cadarache (Mr E. Chojnacki), University of Lappeenranta (Prof. H. Kalli) and PSI of Villigen (Dr N. Aksan) must be acknowledged.

The work of GRS was funded by the German Ministry for Education and Research (BMBF) under contract RS 961.

The work of IPSN has been performed in the frame of a Thermal Hydraulic Group (GTTh), headed by M. Champ. This group is grateful to D. Bestion and G. Geffraye of the CATHARE team for their valuable help.

ENUSA wish to thank Professor Ricardo Bolado of Madrid Polytechnic University for his collaboration and guidance in the ENUSA uncertainty analysis.

Review comments to the comparison report, reference /4/, were received from David Richards and Darryl Dormuth (AECL, Canada), Michel Réocreux (IPSN, France), Alessandro Annunziato (JRC Ispra, Italy), Yutaka Kukita (Nagoya University, Japan), Fernando Pelayo and Rafael Mendizabal (CSN, Spain), and Nusret Aksan (PSI, Switzerland). The authors would like to thank the reviewers for the care they have taken and for their constructive comments. In all 214 comments were provided. These have led to worthwhile improvements to the report.
References


Figure 1: Uncertainty ranges and data for pressure.
Figure 3: Uncertainty ranges and data for hot rod temperature
SAFETY ANALYSIS:
THE TREATMENT OF UNCERTAINTY

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The Utilization of Best Estimate Methodology in Reactor Safety Analysis
OECD/CSNI Seminar
Ankara, Turkey
1998 June 29 - July 1
Outline

Abstract

1. Safety Analysis Objective

2. Role of Validation

3. Structured Approach

4. Dominant Phenomena

5. Treatment of Uncertainties

6. Comparisons to Other Methods

7. Example of Approach

8. Future Plans

9. Conclusion
Safety Analysis: The Treatment of Uncertainty
R.B. Duffey, H.E. Sills, A. Abdul-Razzak, V.S. Krishnan,
B.H. McDonald and T. Andres

Abstract
The use of best estimate safety analysis methods requires a statement of the accuracy of the estimates. The statement of accuracy is inherently linked to the chosen safety analysis methodology. This paper describes a structured approach using best estimate methods considered within an integrated three step process of code, representation, and plant uncertainty assessment. The process is made manageable by ranking important phenomena/parameters and assessing only the most important uncertainty sources. A rationalized bias can also be included to compensate for excluded components of uncertainty. Standard methods are employed in combining uncertainties from identified sources into a statement of overall uncertainties for selected acceptance parameters for the safety analysis.

1. Safety Analysis Objectives
The use of best-estimate safety analysis methods requires a corresponding statement of the accuracy of the estimates. The objectives of uncertainty assessment (UA) for a full scope safety analysis are to:

- provide a statement of the uncertainty in safety margins, utilizing a specified safety analysis method, for a specific plant transient or accident;
- provide an aid for defining plant operating and safety limits;
- devise the appropriate generic methodology and approach to be used to define the statement of uncertainty; and
- provide a scrutable and defensible approach which is consistent with best practice and the limitations of our state of knowledge.

2. The Role of Validation and Best Estimate Methods
Any safety analysis relies on the use of computer codes, physical models, correlations and engineering judgment. Previous approaches to safety analysis include Risk Assessment, as part of waste storage assessments, Probabalistic Safety Analysis, as part of overall plant design and safety review for severe accidents and beyond-design basis events, and Deterministic Safety Analysis, to define plant operating limits for specified transients.

Each approach requires validated methods, which we define as the tools and techniques having specified statements of applicability and accuracy for the specific application, for the selected plant, and for the transient under examination. Validation exercises therefore produce quantified statements of the ranges and associated uncertainties for a specific application and qualify the method for the intended use. Therefore, the statement of accuracy for a particular method is inherently and intimately linked to the chosen safety analysis application.
Where multiple methods, say a linked code suite are used, the validation should focus on the most important, sensitive and/or uncertain parameters.

Since the validation is for specified transients and operating states, the validation must be performed for relevant conditions and designs. This leads directly to the concept of phenomena based validation “matrices” where the method is tested against the ability to predict experimental data, known analytical solutions, or other (numerical and physical) benchmarks.

The output of the validation exercise is a statement of code accuracy and its associated uncertainty for the specified use, including the uncertainties in the ranges of the physical variables and the consequences of the various approximations that may be employed.

In order to conduct a safety analysis, it is necessary to consider the uncertainties in the safety analysis itself, which includes not only the modeling of the important phenomena but also the representations of the physical plant, equipment, or supporting experiments. To conduct this, we invoke a structured approach, adapted from the systematic steps in the Code Scaling, Applicability, and Uncertainty (CSAU) assessment methodology [1] to include the changes relevant to CANDU® reactors, e.g.

1. inclusion of coolant void reactivity feedback during transient conditions;
2. utilization of linked code suites to describe and model the physics, fuel, channel, thermal hydraulics, materials behavior and failure, activity transport and release;
3. definition of plant safety trip and set points, safety system actuation and plant operating state; and
4. utilization of more inclusive and multiple safety parameters.

2.1 Computer Code Validation

The code validation process adopted by the Canadian nuclear industry is based on the five step process illustrated in Figure 1. The first two steps, formation of a Technical Basis Document (TBD) and a Validation Matrix, are generic. The last three steps, establishing a Validation Plan, performing Validation Exercises, and providing a summary Validation Manual, are code version specific.

The TBD identifies safety concerns and accident scenarios. Each accident scenario is subdivided into phases as different phenomena are dominant during different phases of a transient. Within each accident phase, the phenomena are ranked as to primary or secondary importance similar to the Phenomena Importance Ranking Table (PIRT) concept used in the CSAU methodology. The PIRT concept has been extended to rank the important parameters as well. This final step is necessary as uncertainties are evaluated for parameters and the assessment process is made manageable by reducing the number of phenomena/parameters to include only the important ones.

The Validation Matrix identifies data sets that can be used for code validation purposes and cross references these with the important phenomena identified in the TBD.
The Validation Plan describes how the Validation Matrix information is going to be used to estimate the systematic simulation uncertainty in important output parameters for a selected code version for selected accident scenarios. The Validation Exercises record the assessment process and results. The summary Validation Manual provides evidence for the accident scenarios and parameter ranges for which validation has been completed, and the range of uncertainty in important output parameters.

Development of the TBD and Validation Matrices for all safety analysis disciplines has been done as part of an Industry-wide initiative, the Industry Validation Matrices (IVM) [2]. As AECL and the Canadian utilities have differing codes suites for safety analysis, another initiative, the Industry Standard Toolset (IST), is attempting to reduce the number of code versions requiring validation by selecting a single code version or combining the functionality of similar codes.
The exact version of each analysis code have been identified and placed under configuration management and change control. Software quality assurance (SQA) complies with the Canadian Standard N286.7. For so-called legacy codes, which may predate the current standards, a prudent and reasonable review is being performed. Note that developmental code versions cannot be used for safety analysis, but can be of use for sensitivity and confirmatory analyses.

By attempting to minimize the number of methods available and in active use, the cost of safety analysis and software maintenance is reduced. The important technical plus is that different plants and transients are analyzed by different people using similar methods.

These methods therefore represent the “best estimate” codes and methods available for current use in safety analysis.

2.2 Accident Scenarios

The transients of interest which are analogous to the Chapter 15 events in a light water reactor final safety analysis report (LWR FSAR), include the following:

- **Large Break LOCA** - with/without Class IV power - with/without Emergency Core Cooling (ECI);
- **LOCA / LOECl** - large & small breaks - with Class IV - without ECI;
- **small LOCA** (feeder breaks, single channel events, breaks in HTS auxiliary systems - with Class IV - with/without ECI;
- **Loss of Flow** - loss of Class IV power, pump trip, pump seizure, loss of shutdown cooling system pumps;
- **Loss of Regulation** - loss of reactivity control, process control functions (heat transport system (HTS) pressure & inventory control);
- **Loss of feedwater** - all breaks in the feedwater & condensate systems, loss of feedwater pumps and control valve failures; and
- **Steam line break** - includes cases where relief valves fail open.

Some of these scenarios are precursors to more severe accidents and correspond to Level 1 PSA analysis events. They form the basis for the Limiting Conditions of Operation (LCOs) for the plant safety systems, and thus define the Limiting Operating Envelope (LOE).

2.3 Important Analysis Parameters

In general, the safety significant analysis parameters for CANDU include the following fifteen items;
1. Public dose
2. Probability of ROP/NOP trip before dryout for slow Loss of Regulation (LOR) events
3. Peak containment pressure for LOCA
4. Peak containment pressure for steam and feedwater system breaks
5. Minimum containment pressure
6. Peak hydrogen concentration in containment
7. Extent of pressure tube strain for LOCA with ECI
8. Peak fuel sheath temperature for small and large LOCA with ECI, plus all non-LOCA events
9. Peak fuel centre line temperature for small and large LOCA with ECI, plus all non-LOCA events
10. Extent of fuel sheath strain for LOCA with ECI
11. Extent of constrained axial fuel string expansion
12. Extent of calandria tube dryout
13. Subcriticality margin after shutdown
14. Peak pressure in the heat transport system
15. Peak calandria pressure

3. The Structured Approach

The analysis of reactor safety margins utilizing a specific best estimate safety method or code suite proceeds with an integrated three step process, preceded by a statement of the selection of the method. The three steps are:

1. Establishment of Code Uncertainty (CUA)
2. Assessment of the Representation Uncertainty (RUA)
3. Definition of the Plant Uncertainty (PUA)

These main steps in the process (illustrated in Figure 2) are defined as follows.

<table>
<thead>
<tr>
<th>Code Uncertainty</th>
<th>Uncertainty associated with models/correlations, the solution scheme, model options, unmodeled processes, data libraries and/or deficiencies of the computer program.</th>
</tr>
</thead>
<tbody>
<tr>
<td>Representation Uncertainty</td>
<td>Uncertainty in representing/idealizing the real plant such as initial conditions, boundary conditions, plant state, nodalization, scaling (including 3D effects), fabrication tolerances and/or analysis assumptions.</td>
</tr>
<tr>
<td>Plant Uncertainty</td>
<td>Uncertainty in measuring/monitoring the real plant such as reference plant parameters, instrument error, setpoints, instrument response, design allowances and/or availability requirements.</td>
</tr>
</tbody>
</table>
Figure 2  Information and process flow within the structured approach to uncertainty assessment.

3.1  Code Uncertainty Analysis (CUA)

Utilizing specified data and tests from the Industry Validation Matrices (IVM), a code or method is chosen which can provide a statement of the following:

1. the applicability of the method when applied to the phenomena of importance to the specified transient and plant;
2. rationale for the adoption of the particular Code Version or Analysis Method and, placing of those Tools under Configuration Management and Change Control (CM/CC);
3. a statement of the systematic bias and random errors when applied and compared to the chosen test data from the IVM; and
4. a statement of the models/correlations and methods utilized, and their ranging and use within their specified application, and comparison to the stated or original range of applicability.

The output of these analyses are embodied in the Validation Manual for the given plant scenario as follows:

- fatal or non-fatal errors encountered and their resolution
- rationale for adopted model/ noding
- choice of and rationale for the models and correlations for the test case(s)
- range of demonstrated or claimed applicability (transient and reactor state)
- limitations on applicability
- rationale for chosen test comparisons (scaling basis and data quality)
- phenomena identification and ranking for case(s) studied
- routines of code(s) exercised in cases studied (as slave model if appropriate)
- models, correlations and options used and their basis
- noding consistency with, or change from reactor application and its effect
- comparison to physical, engineering or mathematical solutions or correlations
- method and coding used to apply input parameter ranges and uncertainties
- method and coding used to develop accuracy statement
- treatment of combined uncertainties
- run time statistics and sensitivity
- accuracy statement versus data or benchmark, including experimental error, for the transient and steady-state
- recommended corrections/ multipliers to be applied to code variables or inputs
- compensating errors discovered and their correction/ treatment
- quantitative difference from previous code versions and analyses
- sensitivity to material variables(where applicable)
- actual distribution of the assessment results, including 95% confidence limits tabulation of comparisons and data values
- coding and treatment of uncertainty combination (second order and higher) effects
- treatment of propagation of uncertainty/error
For CANDU reactors, some 17 or more codes have been identified as important for safety analysis (listed in Appendix B). For a safety analysis, many of these codes are linked together to form a code suite (essentially a “super” code). The code suite qualification must meet the same requirements specified for single codes.

3.2 Representation Uncertainty Analysis (RUA)

The chosen code or method is then applied to a particular plant (type, design, power, etc). It is necessary to provide a statement of the following:

1. the uncertainty of the method when applied to modeling of the phenomena of importance to the specified transient and plant;
2. rationale for the adoption of the particular nodalization and plant representation scheme;
3. a statement of the systematic bias and random errors incurred in adopting the specific method to the plant scenario;
4. a statement of the correlations and methods utilized, and their ranging and use within their specified plant application and comparison to the stated or original range of applicability; and
5. a statement of the effect of any modeling changes that are outside the range of the validation process.

The output of these analyses are embodied in a Safety Analysis Basis document with specified uncertainties and bias including assessment of the following:

- scaling uncertainty/bias, including 3D effects
- sensitivity to input and model parameter ranges
- sensitivity to reactor state variables (where applicable)
- specified ranges and uncertainties for plant model parameters
- statement of the requirements for given input variables required to complete the analysis
- method and coding used to apply input parameter ranges and uncertainties
- method and coding used to develop accuracy statement
- treatment of combined uncertainties
- actual distribution of the assessment results, including 95% confidence limits tabulation of comparisons and data values
- coding and treatment of uncertainty combination (second order and higher) effects
- treatment of propagation of uncertainty/error
Thus the Plant Model at this stage includes the uncertainties not just in the code(s) or method(s), but also includes uncertainties inherent in the representation of the plant.

3.3 Plant Uncertainty Analysis (PUA)

The output of the plant uncertainty analyses will be included in the Safety Analysis Basis document with a statement of uncertainties and bias including an assessment of the following:

- sensitivity to plant input parameter ranges
- sensitivity to reactor state variables (where applicable)
- specified ranges and uncertainties for plant model parameters, including trip and safety settings
- statement of the requirements for given input variables required to complete the analysis
- method and coding used to apply input parameter ranges and uncertainties
- method and coding used to develop accuracy statement
- treatment of combined uncertainties
- actual distribution of the assessment results, including 95%/95% limits, tabulation of comparisons and data values
- coding and treatment of uncertainty combination (second order and higher) effects

4. Dominant Phenomena

To make the assessment process manageable, the number of phenomena and parameters to be qualified must be reduced to a minimum, but justifiable, number. This reduces the number of computer runs to a manageable set. For example, in the computation of coolant void reactivity, it is estimated that about 200 days of computing would be required for a full unreduced, three-dimensional coupled core transient analysis. Therefore, problem reduction by simplification and importance ranking are necessary.

This ranking process is formally called Importance sampling, and a number of formal techniques are available to help in this key step. The Technical Basis Document (TBD) specifies the minimum number by identifying the important phenomena and key parameters associated with each accident scenario of interest for which uncertainty assessment may be required.

For each scenario, an event sequence description is used in the TBD to identify the phases of the scenario, the governing phenomena, and the key parameters. The Phenomena Key
Parameter Importance Ranking Table (PKPIRT) concept is a formal extension of the Phenomena Important Ranking Table (PIRT) developed for CSAU\(^1\) and forms the basis for identifying the key parameters in the TBD.

This problem reduction (by its very nature) requires some demonstration of the completeness of the prior selection, which may have inadvertently eliminated unknowns or multiple embedded interactions between variables (e.g. the effect of pressure tube creep on coolant void reactivity during a flow transient). Therefore, the ranking and selection must be conducted by experts, and be subject to independent review. In the Ontario Hydro approach, a final check of the rank ordering of importance is done when the sensitivity of the final calculations to uncertainties is determined, and cross-checked with the initial judgments. In CSAU (Reference 1 Part 4) this was achieved by a final check against integral data sets, both in terms of their accuracy and the statistical distribution of the computed final response surface.

In addition, there is always the opportunity to adopt a specific and identified bias, on the basis of rational arguments as to its magnitude and cause, to compensate for additional excluded components of uncertainty.

Thus one can be sure that items originally discarded as unimportant, on the basis of judgment and/or independent sensitivity analysis, are considered in the final result.

5. Treatment of Uncertainties

5.1 Identifying Sources of Uncertainty

Uncertainty assessment is required for each phase of a scenario as the governing phenomena and key parameters are different within each phase, and their specific behaviour may be a consequence of their history.

Uncertainty has two components - systematic and random. To facilitate the uncertainty assessment, all sources of uncertainty that affect the governing phenomena are tabulated, classified as to systematic or random, and their uncertainty range provided.

The contributors to uncertainty are further divided into three generic categories of CUA/RUA/PUA and address the attributes identified in section 3.

To assess the contribution to overall uncertainty in a selected output parameter, the sensitivity (or response) of the analysis to variations in input/model parameter variability on that parameter must be assessed as the contribution to overall uncertainty is directly related to the product of the sensitivity and the magnitude of the input/model parameter

\(^1\) The CSAU methodology does include a process for identification and ranking of Candidate Parameters to obtain Relevant Parameters (see Reference 1, Part 3)
uncertainty. In the final assessment, only those input/model uncertainties that make a significant contribution to overall uncertainty are considered.

5.2 Probability Distribution Functions (PDFs) and Input Response Functions

It is important to note that the methods adopted all include some means to propagate uncertainties throughout the safety calculation. In order to do this in a cost effective manner, subsets of all the possible parameters are chosen based on expert judgment and calculation. The known ranges and distributions of these parameters are input to a set of calculations using the standard toolset (IST) in combination as required. Again methods must be used to reduce the set of calculations to a manageable amount. Consistency checks should be applied when possible.

No a priori judgment need be made about the shape and type of distribution (normal, skewed, uniform). It is prudent to adopt standard techniques where possible, and we prefer to use COTS\(^2\) methods where possible and subject them to benchmarking and standard checks. It is not expected that the mathematical methods will have significant errors compared to the physical uncertainties.

5.3 Combined Uncertainties

A variety of methods have been devised to assess the propagation of errors through an analysis. For linear systems in which the input parameters are independent and the errors are normally distributed, Linear Error Propagation can be used to assess the maximum overall uncertainty in output parameters. In analyses where the error components are small, error estimates can be obtained using perturbation approaches about a reference case [3]. Response surface techniques, coupled with Monte Carlo sampling, are also widely used to estimate the overall uncertainty in complex systems.

In the perturbation methods, variation in a selected output parameter, \(Y\), to variations in input/model parameters, \(X_i\), are based on the total differential representing linearized system response:

\[
\Delta Y = \Sigma \left( \frac{\partial Y}{\partial X_i} \right) \Delta X_i
\]

where the \(\Sigma\) symbol represents a summation over all input uncertainties. If a physically based expression for \(Y\) is available, the expression can be differentiated directly to assess the sensitivities. Adjoint methods [3] can also be used to obtain estimates of the sensitivities about a reference case.

In Linear Error Propagation, the perturbation \(\Delta X_i\) is treated as a random variable, and the variance operator \(V\{\}\) is applied to the equation above to estimate the uncertainty in the output, \(\Delta Y\), according to the expression:

\(^2\text{Commercial Off The Shelf Software, which must itself be subject to common acceptance, known available versions, and readily open to independent checking as required.}\)
\[ V(\Delta Y) = \sum (\frac{\partial Y}{\partial X_i})^2 V(\Delta X_i) \]

where it is assumed that each partial derivative is constant and not uncertain, and that the parameters are all statistically independent. If the perturbations \( \Delta X_i \) are all normally distributed, then \( \Delta Y \) is also normally distributed. The sensitivities, \( (\partial Y/\partial X_i) \), capture system response based either on appropriate data, or from calculation using computer programs or calibrated response functions. The equation for the variance of \( \Delta Y \) can be rewritten in a somewhat more familiar form for the standard deviation of \( Y \), written \( \sigma_Y \):

\[ \sigma_Y = (\sum (\frac{\partial Y}{\partial X_i})^2 \sigma_{X_i}^2)^{1/2} \]

In highly non-linear systems, the original perturbation analysis may require 2nd order or higher terms (including interaction terms), and the partial derivatives themselves may need to be treated as random variables since the sensitivities may change over the uncertainty range of \( X_i \). If the parameters are not statistically independent, the expressions also get much more complicated, with correlations appearing. If the uncertainties in the parameters are not normally distributed, then the uncertainty in \( Y \) may or may not be normally distributed.

5.3.1 Coupled Physical Expressions

A physically based expression may be available to represent the key output parameter of interest. For a simple linearized example, sensible heat to the non-boiling coolant (Q) in an arbitrary volume can be found using the expression:

\[ Q = m \, C_p \, \Delta T \]

where \( m \) is the mass flow rate,

\( C_p \) is the coolant specific heat, and

\( \Delta T \) is the temperature rise in the flow volume of interest.

Thus, we have, in a transient with an incremental linearized time interval, \( \Delta t \), within the transient,

\[ \Delta Q = \left( a_m \frac{\partial m}{\partial t} + a_{\Delta T} \frac{\partial (\Delta T)}{\partial t} \right) \Delta t + a_{m} \frac{\partial m}{\partial t} \frac{\partial (\Delta T)}{\partial t} \Delta t^2 + \ldots \]

The unknown first order coefficients, \( a_\ast \), are derived from a supplementary analysis of the independent time rate of change of the heat input to variations in the flow rate and temperature, and the third term is the cross-product (interaction), which is also assumed to be a linear term (though quadratic in \( \Delta t \)) in this representation. Note that uncertainty in the design values (but not the actual operating conditions) are easily included at this stage.
(e.g. the mass flow rate uncertainty includes the manufacturing (as-built) tolerances in the flow area and losses).

If the uncertainty distributions in \( m \), \( C_p \) and \( \delta T \) are known, several solutions of the above expression can be run where values sampled from these parameter uncertainty distributions are used for each separate solution. The result will a distribution of results for \( Q \) (its uncertainty distribution) which can be analyzed (statistically) to assess the uncertainty components in \( Q \). This is equivalent to the classic treatment of the linear combination of errors of observation in experimental science.

Where the uncertainty distributions for \( m \) and \( \delta T \) are unknown, it will be necessary to successively substitute physical expressions until all the identified uncertainty source parameters (i.e., code, model, and plant) have been incorporated. At that time, uncertainty sampling for a large number of solutions can be carried out to create the output parameter uncertainty distribution.

In the CSAU approach, an equivalent method was used (Reference 1 Part 6), where physically-based correlations were derived, and their uncertainties combined in quadrature as an independent check on the output (response surface) of the computer analyses.

5.3.2 Response Surfaces and Confidence Levels

Creation of coupled physical expressions which incorporate all the identified uncertainty source parameters (see section 5.3.1) are generally not possible to construct. In this case, a surrogate, a response surface, is created by regression fitting the output parameter of interest and corresponding sets of uncertainty source parameters to a generalized polynomial of sufficient order to capture system non-linearities and coupling between input parameters.

The response surface input parameters can then be randomly sampled to create the uncertainty distribution for the selected output (safety-related) parameter, typically using Monte Carlo sampling techniques using random number generators for the sampling methodology, to derive the overall response surface and the statement of confidence levels. We believe that 95% confidence levels are appropriate and sufficient (Reference 1 Part 1), especially when additional biases for unknown or unrepresented effects, or operational limits in the plant state have been included. The extent of sampling, and the method of random sampling are also subject to sensitivity checks.

Existing fitting and sampling software is available (e.g. TableCurve, Excel and SYVAC3 [4]). We expect that the results can be duplicated in the statistically equivalent sense. Thus statistically equivalent statements can be made at the same level of confidence which may have minor differences in magnitude depending on the number of terms included in the polynomial representations of the PDF. If any of the uncertainty source parameters are sufficiently independent of the others (i.e., no parameter coupling or
covariance), then they can be removed from the overall response surface and represented by their own correlation.

6. Comparisons to Other Methods

It is important to state that what is being undertaken is not unique, but simply a practical variation of previously proven concepts, taking advantage of the previous work [3].

A review of four of the differing approaches to the treatment of uncertainties in thermal hydraulic codes has been given by Glaseer and Pochard [5]. There also exists the UMAE approach (Uncertainty Methodology Based on Extrapolation of Accuracy) as given for example in Reference 6 for the analysis of small breaks. In addition, there are substantial bodies of work on post-closure risk assessment for nuclear waste repositories which embody many of the same ideas and techniques [7].

In passing, we note that all approaches basically contain the same elements to meet the same objective, i.e.: to undertake a feasible safety analysis with a reduced parameter set of important input variables, utilizing a selected best estimate theoretical (computer) model representation of a known design configuration with known or estimated uncertainties in the operating state, and including the uncertainties in both the physical and mathematical representation of that design.

The approaches only differ in the relative weight to the experimental, analytical and judgmental determination of uncertainties. Thus, depending on the availability of data and the particular application, alternate rationales were used to reduce or minimize the number of calculational variables, and in choosing the specific methods adopted to develop statements and combinations of uncertainties. There is no completely general theory available. Therefore, it is not a matter of which method is best (or possibly more rigorous), but more which is appropriate and sufficient for the chosen application.

Thus, CSAU [1] is a method which included three elements:

1. requirements and code capabilities;
2. assessment and ranging of parameters; and
3. sensitivity and uncertainty analysis,

which elements are subdivided into 14 specific procedural steps. The LOCA demonstration required 8 runs with one code, with 23 supplementary calculations (totaling 184 data points) for the PDFs, which were then randomly sampled 50,000 times. Finally, five (5) separate biases were added for CUA, RUA and other effects.

The GRS approach was based on random sampling of the analysis input parameter values, and hence needed a minimum of 59 runs with one code to obtain 95% confidence levels. The number of runs was independent of the number of parameters. There was no discussion of, or allowance for, separate biases.

The UMAE approach examined the accuracy of calculations with one code against multiple selected (scaled) experiments for a given transient. From the experimental and theoretical comparison was determined an “average” and “95th percentile uncertainty”,

497
which was then multiplied by an estimate of the (assumed) increase in error with scale [8] which, in effect, serves as an additional experimental scaling bias. This procedure determines a combined (and inseparable) CUA and RUA uncertainty.

By comparison, the Canadian waste risk assessment methodology utilized two scenarios and two system models. For the scenario (actually set of scenarios) representing natural evolution of the disposal environment, 512 simulations were carried out according to a statistical design, with input parameters set at their 1, 50 and 99 percentiles. Sensitivity screening analysis yielded eight (8) influential, eighteen (18) less-influential and 1300 non-influential parameters. In parallel, the input PDFs were sampled randomly 40,000 times, to generate confidence levels on mean radiation dose directly, without the use of a response surface.

The present approach simply combines items from all the above and provides additional specific structure, recalling that CANDU in general requires multiple codes to be used in series for a given analysis (see Figure 3). Thus, in effect:

1. The IST selection of methods includes CSAU Element I, Steps 1, 4 and 5;
2. PKPIRT corresponds to CSAU Element I, Step 3;
3. IVM is CSAU Element 2, Steps 2 and 7;
4. CUA corresponds to CSAU Element 2 Steps 2, 8 and 9;
5. RUA corresponds to CSAU Element 2 Steps 2, 8 (noding) and 10; and

1. PUA includes CSAU Steps 2, 11, 12 and 13.

![Figure 3 Example of a CANDU analysis code suite for reactor transients.](image)

Thus, it is argued that the present approach is indeed a logical and structured methodology for transient safety analysis.
7. Example of Approach

The application of the proposed methodology to the blowdown phase of a large LOCA analysis was conducted. A summary of the main steps and results are given below.

1. The key output parameters related to safety concerns for the blowdown phase of large LOCA are fuel sheath maximum temperature, fuel centerline maximum temperature and pressure tube strain.

2. CATHENA 3.5b rev. 0 was used in the analysis. CATHENA is a one-dimensional, two-fluid thermal hydraulic computer code developed by AECL. It is designed for the analysis of two-phase flow and heat transfer in piping networks. The primary focus of the development has been on the analysis of the sequence of events which occur during a postulated LOCA in a CANDU® reactor.

3. A parameter identification ranking table (PIRT) was generated to rank the contributors to the uncertainty for this event. This included those related to the code, the plant representation and the plant state.

4. Plant operating parameters and accident boundary conditions (PUA), such as break size and location and emergency core cooling system (ECCS) parameters, were set at their conservative values and accordingly were excluded from the analysis. The nodalization scheme of systems is the same as those used in the validation process. Thus there was no additional allowance for RUA.

5. PDF construction and sensitivity analysis on the identified phenomena and modelling parameters were conducted. The analysis was simplified to include only two of the dominant parameters since the objective was the demonstration of the methodology. The most dominant parameters were found to be the fuel thermal conductivity (k) and fuel-to-sheath gap conductance (g), simply because of their impact on the stored energy for the fuel bundle.

6. A response surface PDF, for each output parameter as a function of the change in fuel thermal conductivity and gap conductance, was obtained by fitting second order polynomials. This was selected based on the trends of code response within the uncertainty range of ±2 standard deviation (σ) of k and ±50% of g. The calculation matrix used was:

<table>
<thead>
<tr>
<th>Base case</th>
<th>best estimate k &amp; g values</th>
</tr>
</thead>
<tbody>
<tr>
<td>k</td>
<td>-2σ, -1σ, +1σ, +2σ</td>
</tr>
<tr>
<td>g</td>
<td>-50%, -25%, +25%, +50%</td>
</tr>
<tr>
<td>(k &amp; g)</td>
<td>(-1σ, -25%), (+1σ, +25%)</td>
</tr>
</tbody>
</table>

for a total of 11 CATHENA runs
7. The response surfaces were used to generate the probability distribution function of the output variables by using 10,000 random combinations of thermal conductivity and gap conductance.

8. The calculated uncertainty in the output parameters with confidence level of 95% were obtained from the calculated probability distribution function of the output variables.

As an example, the probability and cumulative distributions of the resultant sheath temperature, where thermal conductivity and gap conductance values are drawn from normal and uniform distributions respectively, are shown in Figure 4.

![Graph showing frequency and cumulative distribution of maximum sheath temperature](image)

**Figure 4** Frequency Response and Cumulative Distribution of Maximum Sheath Temperature

Table 1 shows various values of the output variables. The column marked “Response Surface” shows the 95% one-sided confidence bound. The column labeled “Linear Error Propagation” shows values that should be about 2 standard deviations above the mean because they were obtained by summing in quadrature the 2 sigma variations in each of the input parameters.
<table>
<thead>
<tr>
<th>Output Parameter</th>
<th>Base Case</th>
<th>Linear Error Propagation</th>
<th>Simultaneous Combination</th>
<th>Response Surface</th>
</tr>
</thead>
<tbody>
<tr>
<td>Sheath Maximum Temperature (°C)</td>
<td>1175</td>
<td>1296</td>
<td>1346</td>
<td>1272</td>
</tr>
<tr>
<td>Fuel Centrel ine Maximum Temperature (°C)</td>
<td>2025</td>
<td>2384</td>
<td>2498</td>
<td>2320</td>
</tr>
<tr>
<td>Channel L1 Pressure Tube Maximum Strain (%)</td>
<td>0.507</td>
<td>1.24</td>
<td>1.544</td>
<td>1.05</td>
</tr>
</tbody>
</table>

**Table 1 Comparison of the Various Methodologies for Combining Uncertainties**

The “Simultaneous Combination” column of Table 1 was obtained from a single run of the model which simultaneously set the parameters to values 2 sigma above their means. This type of analysis, based on perturbation analysis, seems like a reasonable way of setting an upper bound. In fact it is not dependable, because it does not take into account the variation in the model. In these examples, the value so obtained is conservatively high, because the model is monotonic in each parameter (i.e., the output goes up whenever the parameter goes up). If instead the output variable had a quadratic dependence on one of the parameters (i.e., with a peak value somewhere inside the range), this estimate could just as easily be low.

The sensitivity to distribution functions of the input variables (fuel thermal conductivity and gap conductance) were checked by considering three different combinations of distributions. Both normal, both uniform and a combination of normal distribution for thermal conductivity and uniform distribution for gap conductance were considered. The results are shown in Table 2. The process of fitting response surfaces, random value generation, and generating the probability distribution function of the output variables were conducted using EXCEL. The goodness of the fitted response surface was checked by fitting large number of linear and non-linear relations available in TableCurve for sheath temperature. The results of the best fitted curve by TableCurve resulted in only a 1.5 °C difference in the predicted maximum sheath temperature at 95% confidence level when compared to that obtained from the polynomial fit obtained by EXCEL.
Table 2 Sensitivity to Parameter Distribution Functions

<table>
<thead>
<tr>
<th>Output</th>
<th>Both uniform</th>
<th>Normal Thermal Conductivity &amp; uniform Gap Conductance</th>
<th>Both Normal</th>
</tr>
</thead>
<tbody>
<tr>
<td>Sheath Maximum Temperature (°C)</td>
<td>1281</td>
<td>1272</td>
<td>1268</td>
</tr>
<tr>
<td>Centreline Maximum Temperature (°C)</td>
<td>2350</td>
<td>2320</td>
<td>2308</td>
</tr>
<tr>
<td>Channel L1 Pressure Tube Maximum Strain (%)</td>
<td>1.09</td>
<td>1.05</td>
<td>1.03</td>
</tr>
</tbody>
</table>

The above analysis demonstrates the effectiveness of using a structured approach to response surface development, and the insensitivity of the results to some assumptions.

For CANDU designs and selected operating units, AECL intends to conduct the following:

1. Completion of the demonstration of the large LOCA analysis, using more complete data and information for the RUA and PUA uncertainty elements;
2. Complete the IVT exercise, thus providing additional input on the CUA contributors;
3. Extension of the analysis to Loss-of-Flow for CANDU® plants, and to small breaks; and
4. Utilizing uncertainty analysis to refine safety margins where it is prudent and cost-effective to do so, particularly in the matter of the LCO’s, operating procedures, and design limits for CPR, CVR and plant trip settings.

9. CONCLUSION

This paper describes a structured approach using best estimate methods considered within an integrated three step process of code, representation, and plant uncertainty assessment. The statement of accuracy is inherently linked to the chosen safety analysis application.

The process is made manageable by ranking important phenomena/parameters and assessing only the most important uncertainty sources. A rationalized bias can also be included to compensate for excluded components of uncertainty.

Standard methods are employed in combining uncertainties from identified sources into a statement of overall uncertainties for selected acceptance parameters for the safety analysis.
References:


Appendix B

**Industry Standard Tools**

For AECL-designed CANDU® reactors, a set of methods have been developed which are reasonably consistent across the operating plants and new designs. These are listed below:

<table>
<thead>
<tr>
<th>Computer Program</th>
<th>Usage</th>
</tr>
</thead>
<tbody>
<tr>
<td>ASSERT-PV</td>
<td>fuel bundle and fuel channel subchannel thermal hydraulics</td>
</tr>
<tr>
<td>CANSIM</td>
<td>coupled ELOCA fuel element / ASSERT-PV channel thermal hydraulics</td>
</tr>
<tr>
<td>CATHENA</td>
<td>system thermal hydraulics</td>
</tr>
<tr>
<td>CATHENA-ELOCA</td>
<td>coupled fuel element / slave channel thermal hydraulics</td>
</tr>
<tr>
<td>CHMWRK</td>
<td>primary heat transport system chemistry control</td>
</tr>
<tr>
<td>ELESTRES IST</td>
<td>fuel element behaviour under normal operating conditions</td>
</tr>
<tr>
<td>ELOCA</td>
<td>fuel element behaviour for high temperature transients</td>
</tr>
<tr>
<td>GOTHIC</td>
<td>containment hydraulics and fluid flow</td>
</tr>
<tr>
<td>MODTURC-CLAS</td>
<td>moderator heat transport</td>
</tr>
<tr>
<td>NEWPEAR</td>
<td>dose assessment from airborne contamination</td>
</tr>
<tr>
<td>NUCIRC</td>
<td>primary system flow</td>
</tr>
<tr>
<td>RFSP/PCM</td>
<td>core wide physics and fuel management</td>
</tr>
<tr>
<td>ROVER-F</td>
<td>ROP setpoint analysis</td>
</tr>
<tr>
<td>SMART</td>
<td>fission product transport in containment</td>
</tr>
<tr>
<td>SOURCE</td>
<td>fission product source term</td>
</tr>
<tr>
<td>TUBRUPT</td>
<td>consequence of pressure tube failure</td>
</tr>
<tr>
<td>WIMS-AECL</td>
<td>lattice cell assessment</td>
</tr>
</tbody>
</table>
SESSION IV:

SELECTED ISSUES OF THERMAL-HYDRAULIC SAFETY ANALYSES
A CASE STUDY ON SMALL BREAK LOCA

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1. Introduction

Turkey is planning to launch a nuclear power program and a decision for the vendor is to be given by the end of 1998. The Turkish Atomic Energy Authority (TAEA), as a regulatory body, requires proposed nuclear power plant be licensable in home country. The principle of licensibility in vendor’s home country is one of the governing factor for the general methodology of reviewing safety analysis report concerning applicable codes, standards and regulations, and methodology for safety analysis as well. In other words, the application of best estimate computer codes with conservative initial and boundary conditions may vary from country to country, however, TAEA, in principle, follows the methodology applied by vendor by the condition that the methodology complies with the regulations of vendor’s country.

In this paper, a case study on a 40 cm² break in the hot-leg injection line of the emergency core cooling and residual heat removal system of a typical PWR is presented and results obtained from the conservative and realistic approaches, with respect to initial and boundary conditions, are discussed.

2. General Characteristics of the Case Study

The accident scenario selected for this paper is applied to a RELAP5 model of a PWR (1000 MWₑ) type nuclear power plant. The break in the hot-leg injection line of the
emergency core cooling and residual heat removal system is assumed to occur between the check valve immediately upstream of the reactor coolant system and the inlet nozzle to the reactor coolant line (Figure 1). This means that loss of coolant can not be prevented by closure of the Emergency Core Cooling System (ECCS) check valves.

The boundary conditions of the case analyzed are as follows:

- Initiation of emergency power mode simultaneous with reactor and turbine trip after onset of the accident
- One emergency power diesel set is down for repair
- One emergency power diesel set fails to start (single failure)
- Break in ECCS injection line connected to operable emergency power diesel set.

A loss of coolant accident with the aforementioned boundary conditions constitutes a special case with respect to accident sequence, system availability and probability. It is clear from these boundary conditions that only one safety injection pump out of three is available for coolant loss make-up as the result of postulated failure and repair of two emergency diesel generating sets. Furthermore it is also postulated that the break at the Reactor Coolant System (RCS) boundary occurs in the ECCS train in which operable High Pressure Safety Injection System (HPIS) is connected. Consequently HP safety injection is severely impaired if not completely prevented. This primarily depends on the break size selected, i.e. although a larger break size, such as circumferential rupture of the injection line, prevents injection of the SI pump completely, the pressure and level in the RCS fall faster so that low pressure injection system cuts in at early stage of the accident and impermissible core uncover is prevented. However, smaller breaks in the ECCS injection line can be made up by the SI pump after a short accident duration without the reduced RCS level causing the cladding temperature to rise. In this type of accident, the pressure remains almost constant in the subsequent course of the event and decay heat is removed by two-phase natural circulation to the secondary side. Larger breaks up to 40 cm$^2$ are special cases in that the RCS pressure drops relatively slowly and the emergency coolant delivered by the SI pump can not enter the reactor system. Instead, since the emergency coolant mixes with reactor coolant upstream of the break location, it has the effect of
subcooling the break mass flow, which results in increased loss of coolant from the reactor coolant system.

3. Boundary Conditions of the Analyses by RELAP5 Code

Two analyses were performed by using the RELAP5/mod3.2 thermalhydraulic computer code:

1- Realistic (best-estimate) analysis:

- Power: 100%
- Hot channel factor: 2.5
- Core configuration: BOC
- MS-RV actuation: 10 minutes
- Rate of MS-RV automatic cooldown: 100 K/h (and optionally 130 K/h)

2- Conservative analysis:

- Power: 106%
- Hot channel factor: 2.5
- Core configuration: BOC
- MS-RV actuation: 30 minutes
- Rate of MS-RV automatic cooldown: 100 K/h

Primary concern of this study is to investigate the effect of time of Main Steam Relief Valve (MS-RV) actuation and rate of MS-RV automatic cooldown. Once the LOCA alarm is given by the signal:

Primary Side Pressure < 132 bar
++
Containment Pressure > 30 mbar
then automatic cooldown is started by the operator after a predetermined period, i.e. 10 or 30 minutes in realistic and conservative approaches, respectively.

4. Results and Discussion

Immediately after onset of the accident, i.e. by opening of the break in the ECCS injection line, reactor and turbine trips are initiated in response of the pressure increase in the containment and the pressure drop in the RCS. The LOCA alarm is given to alert the operator to start the steam generator secondary side cooldown via MS-RVs. Since the emergency power mode is postulated in this analysis, the main steam bypass station is not available and due to this reason the secondary side pressure rises until the safety valves open at about 350 s (Figure 2). The safety injection pump starts to inject emergency coolant at a system pressure of 110 bar and emergency coolant flows directly to the break location where it mixes with reactor coolant thus causing an increase in the mass flow rate from the break due to subcooling effect. In Figure 2 it can be clearly observed that the RCS pressure is governed by that of the secondary side after the actuation of the MS-RVs and can be lowered at the rate corresponding to a rate of 100 K/h. The depressurization of the primary system, via MS-RVs, is initiated at 600 s and 1800 s corresponding to best-estimate and conservative methods, respectively. As can be seen in Figure 3, the peak mass flow rates, after the opening of the MS-RVs by manual action of operator, occur with a delay of 1200 s in two methods. It is interesting to note that while primary pressure strictly follows the trend of the secondary side pressure in best estimate model, the primary pressure predicted in the conservative model falls below the secondary pressure, much before the opening of the MS-RVs, due to dominated energy discharge through the break. This means that heat is transferred from secondary to primary side after 1400 s, in the respective model.

The blowdown rate is characterized by two components: mass flow rate from the primary side and mass flow rate from the SI pump. The mass flow rate to the containment (break flow rate) is always greater than that of flow rate from the RCS, after 800 s (Figure 4). The difference in mass flow rate corresponds to the emergency coolant injected by the SI pump which means that the emergency coolant can not reach to the RCS.
By the time approximately 1250–1500 s have elapsed, the collapsed level in the core has fallen to the mid-core elevation (Figure 5) and the upper regions of the fuel rods are uncovered and are predominantly cooled by the steam evaporated from the coolant remaining in the lower region of the reactor pressure vessel. The collapsed levels corresponding to the best-estimate and conservative models differ after 1250 s and the level predicted by the conservative model is lower than that of best-estimate model. The difference is as high as 0.5 m. These degraded cooling conditions result in a level dependent increase in the core temperatures starting from the upper region. The cladding temperature trends are presented in Figure 6. The conservative model yields a drastic increase in cladding temperature and crosses the safety acceptance limit of 1473 K at about 2100 s. The peak cladding temperature, however, remains below the safety acceptance limit in best-estimate analysis. The sharp decrease in cladding temperature after 3500 s is due to accumulator injection which is enabled due to early (10 minutes) operator action for the MS-RVs in the best-estimate model. This is also apparent from the core collapsed level (Figure 5), i.e. the level drastically increases after 3250 s and reaches up to the upper edge of core at 3750 s. The analysis is terminated by the actuation of Residual Heat Removal (RHR) pump when RCS pressure is 10 bar since emergency coolant from accumulators and RHR system refill the reactor coolant system and the break is closed by the hot-leg injection component.

The effect of the cooldown rate of the MS-RVs is also analyzed by repeating the best-estimate simulation with the cooldown rate of 130 K/h, instead of 100 K/h. The cooldown rate of 100 K/h yields 30% higher peak cladding temperature compared to the case with 130 K/h (Figure 7). This clearly shows that an uncertainty associated with the cooldown rate of MS-RVs should be considered in performing best-estimate analysis.

Another uncertainty to be considered is the critical flow model in the RELAP5 code. Two versions of the RELAP5 code, i.e. mod3.2 and mod3.2.1.2, were used to run the conservative case to see whether the improvement in the Henry-Fauske critical flow model in mod3.2.1.2 effects the cladding temperature trend. As seen in Figure 8, the cladding temperature as predicted by mod3.2.1.2 crosses the safety acceptance limit of 1473 K about 250 s earlier than that of mod3.2. This is simply due to larger break mass
flow rate during the first 800 s of transient which is the result of modified Henry-Fauske model.

5. Conclusion

The accident scenario concerning 40 cm$^2$ in the ECCS injection line has been experienced by the TAEA in early 80s during negotiations by the vendor and is a special case with respect to probability of the boundary conditions, system availability and accident sequence. The importance of this special case lies in the fact that the decision maker (the regulatory body) should decide whether the boundary conditions (timing of the actuation of MS-RVs by operator, cooldown rate of the MS-RVs, initial reactor power) for deterministic analysis could be modified, in the expense of less conservatism, when the probability of the accident scenario is low enough. The uncertainties become very important when performing best-estimate thermal-hydraulic analysis and the decision maker should also consider uncertainties associated with the realistic approach, otherwise an analysis will give no-or insufficient- feeling for the risks of the accident. In the present accident scenario, cooldown rate of the MS-RVs and critical flow model (break flow rate) are the possible sources of the uncertainties which can effect the peak cladding temperature and timing of violation of the safety acceptance limit, respectively.
Figure 4 The Break Location

PWR-1000 40 cm² Break in the ECC Line

Figure 2 The Pressure of Primary and Secondary Sides
Figure 3 The Mass Flow Rate of MS-RVs

Figure 4 The Break Flow Rate
Figure 5 The Core Collapsed Level

Figure 6 The Cladding Temperature
Figure 7 The Comparison of Cladding Temperature for 100 K/h and 130 K/h MS-RV Cooldown Rates

Figure 8 The Comparison of R5/m3.2 and R5/m3.2.1.2 Predictions for Cladding Temperature
A CASE STUDY ON SMALL BREAK LOCA

ALİ TANRIKUT
TURKISH ATOMIC ENERGY AUTHORITY

OECD/NEA SEMINAR ON BE METHODS IN T/H SAFETY ANALYSIS
Ankara, Turkey
29 June - 1 July 1998
CONTENT

• Background of the case study

• General characteristics of the accident scenario (40 cm$^2$ break at the ECCS line)

• Boundary conditions of the analyses by RELAP5

• Results and discussion
BACKGROUND OF THE CASE STUDY

- The code application is based on a PSAR data of a PWR-1000
- TAEA followed the methodology of the vendor country for accident analysis
- The case selected is:
  40 cm$^2$ break at the hot leg injection pipe of ECCS (10% of $A_{\text{total}}$)
  + loss of offsite power
  + safety injection pump injects to the broken ECCS pipe
  + operator action for MS-RV actuation (SG cooldown)
Fig. 1  Break Location and Injection Point of SI Pump
GENERAL CHARACTERISTICS OF THE SMALL BREAK LOCA (40 cm²)

- A special case with respect to

  ⇒ accident sequence

  ⇒ system availability

  ⇒ probability
GENERAL CHARACTERISTICS OF THE SMALL BREAK LOCA (40 cm²) (continued)

Probability:

• The boundary conditions of the accident:
  - Initiation of the emergency power mode simultaneously with reactor and turbine trip after onset of the accident
  - 1 emergency power diesel set is down for repair
  - 1 emergency power diesel set fails to start (single failure)
  - break in injection line connected to operable emergency power diesel set
GENERAL CHARACTERISTICS OF THE SMALL BREAK LOCA (40 cm²) (continued)

System Availability:

- ECCS train available is connected to the pipe line where break is opened
- Safety Injection Pump injects to the break location

⇒ Safety injection is severely impaired if not completely prevented due to break size (40 cm²) and location imposed to the analysis

- Accumulators are available
GENERAL CHARACTERISTICS OF THE SMALL BREAK LOCA (40 cm²) (continued)

Accident Sequence:

- Reactor and turbine trip is initiated ($P_{\text{prim}}$: low, $P_{\text{cont}}$: high)
- MS-SVs regulates the secondary pressure
- SI pump starts to deliver coolant
  - $\Rightarrow$ injected coolant mixes with coolant escaping from primary system
  - $\Rightarrow$ not capable to recover the coolant lost
- Operator action for SG secondary side cooldown via MS-RVs
  - $\Rightarrow$ to lower pressure for accumulator injection
BOUNDARY CONDITIONS OF THE ANALYSES BY RELAP5

Two analyses are performed with different boundary conditions:

- **Realistic (best-estimate) analysis:**
  - Power: 100%
  - Hot channel factor: 2.5
  - Core configuration: BOC
  - MS-RV actuation: 10 minutes
  - Rate of MS-RV automatic cooldown: 100 K/h (and optionally 130 K/h)

- **Conservative analysis:**
  - Power: 106%
  - Hot channel factor: 2.5
  - Core configuration: BOC
  - MS-RV actuation: 30 minutes
  - Rate of MS-RV automatic cooldown: 100 K/h
RESULTS AND DISCUSSION

- The analyses are highly depending on timing of MS-RV actuation by operator

\[ \Rightarrow \textit{realistic} \text{ approach with } t=10 \text{ minutes yields enough depressurization of primary system to ensure clad temperature not to exceed the safety limit (1200 °C)} \]

\[ \Rightarrow \textit{conservative} \text{ approach with } t=30 \text{ minutes results in clad temperature exceeding safety limit (1200 °C) and not permissible from licensing point of view} \]

- Cooldown rate of MS-RV is the source of uncertainty for this analysis

\[ \Rightarrow \text{Even if } t=10 \text{ minutes is taken for MS-RV actuation } 100 \text{ K/h rate yields } 30 \% \text{ higher clad temperature than that of } 130 \text{ K/h} \]
RESULTS AND DISCUSSION (continued)

- The decision maker should decide whether the boundary conditions for deterministic analysis could be modified, in the expense of less conservatism, when the probability of the accident scenario is low enough

  ⇒ e.g. timing of MS-RV actuation by operator, initial reactor power ...

- The decision maker should consider uncertainties associated with the realistic approach, otherwise an analysis will give no feeling for the risks of the accident

  ⇒ e.g. cooldown rate of MS-RVs, break flow rate ...
PWR–1000 40 cm2 Break in the ECC Line

- Primary side (best–estimate)
- Secondary side (best–estimate)
- Primary side (conservative)
- Secondary side (conservative)
PWR–1000 40 cm² Break in the ECC Line

- **clad temp. (best–estimate)**
- **clad temp. (conservative)**

Clad Temperature (K)

Time (s)
PWR-1000 40 cm² Break in the ECC Line

- Solid line: level (best estimate)
- Dashed line: level (conservative)

Core Collapsed Level (m) vs. Time (s)
PWR-1000 40 cm² Break in the ECC Line

Flow Rate (kg/s)

- from primary side (best-estimate)
- to the containment (best-estimate)
- from primary side (conservative)
- to the containment (conservative)

Time (s)
PWR-1000 40 cm² Break in the ECC Line

Comparison of MS-RV Cooldown Rate

- clad temp. (MS-RV 100 K/h)
- clad temp. (MS-RV 130 K/h)

Clad Temperature (K)

Time (s)
Performance Evaluation of the Emergency Core Cooling System for an Evolutionary Pressurized Water Reactor

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Abstract

The emergency core cooling system (ECCS) arrangements to mitigate a loss of coolant accident (LOCA) are studied to support the design of an evolutionary pressurized water reactor. Performance of the ECCS for various cases is evaluated by calculating thermal-hydraulic transients following a large break loss of coolant accident (LBLOCA). The RELAP5/MOD3/K code is used which was improved from the RELAP5/MOD3.1 code and merged with the CONTEMPT4/MOD5 code. Limiting values of initial and boundary conditions are assumed to yield adverse cladding temperature. The ECCS features of a base case study are characterized by four mechanical trains of the safety injection system (SIS) and use of direct vessel injection (DVI) above the cold leg nozzle. Each train of the SIS consists of a high pressure safety injection (HPSI) pump and a safety injection tank (SIT) without a low pressure safety injection (LPSI) pump. The RELAP5/MOD3/K analysis results show that the design criteria of the ECCS cannot be met for the base case study. This result is associated with emergency core cooling (ECC) water bypass to the break and resultant shortage of penetration into the downcomer during the reflood phase after the SITs empty. In order to resolve this problem, various ECCS arrangements such as elevation of the DVI nozzle, injection flowrate increase, and cold leg injection are investigated to obtain an effective ECCS arrangement. The DVI nozzle below the cold leg nozzle is found to be a desired location for an effective ECCS configuration. It is also found that increasing the injection flowrate after the SITs empty is required to enhance the ECCS performance.

Introduction

Various ECCS arrangements have been studied to enhance safety following a postulated LOCA. Several advanced ECCS design features suggested for an evolutionary light water reactor are investigated in the present study. The SIS of four mechanical trains is one of the top-tier requirements to reduce the core damage frequency for the evolutionary pressurized water reactor [1]. The four-train SIS can also result in a simplification of the pump discharge branch lines. Another important requirement is the use of DVI rather than cold leg injection (CLI) to reduce the SIS pump capacity.
required for a cold leg break, and to increase substantially the reliability of safety injection. In order to address the above utility requirements, a SIS configuration consisting of the four-train HPSI pump and DVI is taken as the base case of the present study. It is, however, noted that results from the upper plenum test facility (UPTF) tests [2,3] indicated the use of DVI could reduce effectiveness of ECC injection although the effect of reduced penetration on the peak cladding temperature was less than 100 °F [1]. Therefore, the thermal-hydraulic phenomena associated with DVI are investigated in detail. Since the base case analysis shows unsatisfactory results, additional calculations are performed to obtain effective ECCS arrangements by lowering the DVI nozzle below the cold leg nozzle. Performance of the CLI with the LPSI pump is also evaluated.

Analysis Methodology

Reflood models that play an important role in a LBLOCA analysis has been implemented in the RELAP5/MOD3.1 code [4]. However, assessment calculations of the models are relatively scarce because the RELAP5 code development has been focused on the applications to a small break LOCA analysis. The containment pressure behavior is also important since it is associated with the steam binding in the reactor vessel upper plenum during the reflood phase. However, no specific models for the containment were developed in RELAP5. For more accurate LBLOCA analysis, the RELAP5/MOD3/K code [5] developed by modifying the reflood models of RELAP5/MOD3.1 and merging with the CONTEMPT4/MOD5 code [6] for the containment back pressure is employed in the present study.

A RELAP5 nodalization for LBLOCA analysis is shown in Fig. 1. The reactor vessel downcomer annulus is divided azimuthally into six channels connecting neighboring nodes at the same elevation by cross-junctions to predict the multi-dimensional ECC bypass phenomenon. Flow energy loss coefficients at cross-junctions are assumed to be zero. The countercurrent flow limit option is not employed. Discharge coefficients at break junctions are set to 1.0. The effect of noncondensable gas in the SIT is not modeled by assuming that the connecting valves are closed just before the SITs empty.
Results and Discussions

The important initial and boundary conditions for the analyses are summarized in Table 1. The conservative values, which are identical with those for design basis event analyses performed by the evaluation model, are employed. A single failure concurrent with a LOCA is also introduced. The ECCS arrangements analyzed in the present study are also summarized in Table 2. The calculation results are shown in Fig. 2 to Fig. 5. Major output parameters such as integrated break mass, collapsed water level of the downcomer and the core, and fuel cladding temperature are compared in those Figures.

Base Case

The ECCS of the base case (Case A) consists of four mechanical trains and two electrical trains of the SIS. Two HPSI pumps and four SITs are available since a single failure assumption of a diesel generator would result in the failure of two mechanical SIS trains. The calculated important events following the break are as follows: the SIT injection starts at 16 seconds, the end of blowdown (EOB) occurs at 30 seconds, the start of core reflood occurs at 38 seconds, HPSI pumps start at 54 seconds, and the SITs empty at 73 seconds. However, the code fails at about 430 seconds due to excessive increase of the cladding temperature.

The calculated thermal-hydraulic behaviors before the SITs empty are similar to the typical LBLOCA analysis results. The ECC water injected is sufficient to fill the downcomer annulus to the DVI nozzle elevation. The core collapsed liquid level increases continuously to the mid-height of the core. Accordingly, adiabatic heat-up of the cladding during the refill phase turns around and starts to decrease. After the SITs empty, however, the calculated thermal-hydraulic behaviors are changed drastically. The downcomer collapsed liquid level decreases rapidly below the cold leg, and eventually, the core is uncovered completely. The core uncovery results in the heat-up of the cladding and the code fails eventually.

The water level depression of the downcomer during reflood phase was observed in the UPTF tests [2,3]. The level depression was caused by bypass due to the steam flow of the ECC water injected into the break. In the present calculation, the ECC bypass is too vigorous to deliver the injected water into the core. The steam generated in the core is
superheated and accelerated along the steam generator U-tubes due to heat transfer from the steam generator secondary side. The superheated steam enhances the ECC bypass. The excessive bypass after the SITs empty seems to be partially caused by the conservative initial conditions and/or the insufficient injection flowrate. It should be noted that the one-dimensional code such as RELAP5/MOD3/K can only simulate the overall trends of the ECC bypass using the split downcomer model. For detailed analysis of the ECC bypass, further study using a multi-dimensional thermal-hydraulic computer code is required.

*Elevation of DVI Nozzles*

A sensitivity calculation on the DVI nozzle location was performed to evaluate the effectiveness of DVI nozzles at a higher elevation than the cold leg. The ECC water bypass consisted of carry-over of falling water from higher elevations and entrainment of the downcomer water by the steam flow from the intact cold leg to the break. The carry-over can be eliminated by lowering the injection nozzle below the cold leg (Case B). In this case, the gradient of the integrated break flowrate after the SITs empty, as shown in Fig. 2, is reduced because of elimination of the carry-over. However, the fuel cladding is not rewetted since the core collapsed liquid level is maintained at the mid-height of the core as shown in Figs. 4 and 5. A spin-off of this analysis is to evaluate appropriateness of the injection flowrate after the SITs empty. The calculation results show that the injection flowrate is insufficient to rewet the fuel cladding although the water penetration into the core is more effective than the base case. The results show that an increase in the injection flowrate after the SITs empty is required to enhance the ECCS performance.

*Safety Injection Flowrate*

A sensitivity study on the injection flowrate consists of two cases (Case C and D) with a LPSI pump as shown in Table 2. However, the flowrate increases only 200 kg/s over the base case since two mechanical trains of the SIS are assumed. The calculation results are similar to those of the base case before the SITs empty. The downcomer collapsed liquid level is higher than that of the relevant reference case (Case A to Case C and Case B to Case D), although the break flowrate is also increased after the SITs empty. The reduction of the downcomer level depression results in an earlier reflooding than the reference case. The fuel cladding is not quenched until 500 seconds in Case C, while
the fuel cladding is completely rewetted within about 80 seconds in Case D alluding the effectiveness of lower DVI nozzle.

**Cold Leg Injection**

To investigate differences between DVI nozzles and CLI nozzles, a calculation (Case E) is performed. The CLI with the SIS configuration of the base case is hard to satisfy the design criteria since the ECC water injected at the broken cold leg is spilled to the containment. In Case E, the LPSI pump is added to the base case. The injection flowrate after the SITs empty is identical to that of Case C and Case D because a half of the water injected from pumps available is spilled. Comparisons of Case E to Case C and Case D, as shown in Figs. 2 to 5, show more effectiveness of the CLI nozzle than the upper DVI nozzle but less effectiveness than the lower DVI nozzle in view of timing of the fuel cladding quenching.

**Conclusions**

To obtain effective ECCS arrangements for an evolutionary pressurized water reactor, a sensitivity study for various ECCS arrangements was performed using the RELAP5/MOD3/K code under assumptions of conservative initial and boundary conditions. The following conclusions are drawn from the calculation results.

1. A four-train SIS configuration consisting of DVI and HPSI pumps may not be sufficient to meet the ECCS design criteria. The result is quite associated with both the injection flowrate and the downcomer water level depression after the SITs empty.

2. Based on the sensitivity study for various ECCS arrangements, it is concluded that both lowering the injection nozzle below the cold leg and increasing the injection flowrate after the SITs empty are required to enhance the ECCS performance.
References


Table 1. Initial and boundary conditions for ECCS performance analysis

<table>
<thead>
<tr>
<th>Parameters</th>
<th>Values</th>
</tr>
</thead>
<tbody>
<tr>
<td><strong>Power</strong></td>
<td></td>
</tr>
<tr>
<td>NSSS thermal power, MW</td>
<td>4000</td>
</tr>
<tr>
<td>Peak linear heat generation rate, W/cm</td>
<td>450</td>
</tr>
<tr>
<td>Axial power shape</td>
<td>65% top-to-skewed</td>
</tr>
<tr>
<td>Decay heat model</td>
<td>1.2 * ANS 71</td>
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<tr>
<td><strong>RCS primary</strong></td>
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<tr>
<td>Pressure, bar</td>
<td>155.1</td>
</tr>
<tr>
<td>Temperature at hot / cold leg, K</td>
<td>597.0 / 563.7</td>
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<tr>
<td>Core flowrate, kg/s</td>
<td>20990</td>
</tr>
<tr>
<td><strong>Steam generator</strong></td>
<td></td>
</tr>
<tr>
<td>Tube plugging, %</td>
<td>10</td>
</tr>
<tr>
<td>Pressure, bar</td>
<td>68.95</td>
</tr>
<tr>
<td><strong>Safety injection system</strong></td>
<td></td>
</tr>
<tr>
<td>Flowrate of a HPSI / LPSI pump, kg/s</td>
<td>60 / 260</td>
</tr>
<tr>
<td>SIT pressure, bar</td>
<td>40.3</td>
</tr>
<tr>
<td>SIT gas / liquid volume, m³</td>
<td>22.7 / 45.3</td>
</tr>
<tr>
<td>Outside / inside diameter of DVI nozzle, m</td>
<td>0.3048/0.2159</td>
</tr>
<tr>
<td>DVI nozzle elevation from cold leg centerline (upper / lower), m</td>
<td>2.1 / -0.69</td>
</tr>
<tr>
<td><strong>Containment</strong></td>
<td></td>
</tr>
<tr>
<td>Net free volume, m³</td>
<td>93445.6</td>
</tr>
<tr>
<td>Pressure, bar</td>
<td>1.0</td>
</tr>
<tr>
<td>Flowrate of a spray pump, kg/s</td>
<td>315.15</td>
</tr>
</tbody>
</table>

Table 2. ECCS arrangements of cases analyzed

<table>
<thead>
<tr>
<th>Case</th>
<th>Case A</th>
<th>Case B</th>
<th>Case C</th>
<th>Case D</th>
<th>Case E</th>
</tr>
</thead>
<tbody>
<tr>
<td>Safety injection nozzle</td>
<td>DVI (Upper)</td>
<td>DVI (Lower)</td>
<td>DVI (Upper)</td>
<td>DVI (Lower)</td>
<td>CLI</td>
</tr>
<tr>
<td>SI pump available</td>
<td>2 HPSI</td>
<td>2 HPSI</td>
<td>1 HPSI</td>
<td>1 HPSI</td>
<td>2 HPSI</td>
</tr>
<tr>
<td>Total pump flowrate, kg/s</td>
<td>120</td>
<td>120</td>
<td>320</td>
<td>320</td>
<td>640</td>
</tr>
<tr>
<td>No. of SIT available</td>
<td>4</td>
<td>4</td>
<td>4</td>
<td>4</td>
<td>4</td>
</tr>
</tbody>
</table>
Fig. 1  RELAP5/MOD3 Nodalization for LBLOCA Analysis
Fig. 2  Integrated break mass flowrate

Fig. 3  Collapsed liquid level of the downcomer

Fig. 4  Collapsed liquid level of the core

Fig. 5  Fuel cladding temperature at 3.05m from the active bottom
PERFORMANCE EVALUATION OF THE EMERGENCY CORE COOLING SYSTEM FOR AN EVOLUTIONARY PRESSURIZED WATER REACTOR

July 1, 1998
Presentation in OECD/NEA/CSNI Seminar on
"Best Estimate Methods in Thermal-Hydraulic Safety Analysis"

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REACTOR DEVELOPMENT IN KOREA

- Status of Nuclear Power Plant
  - 12 PWRs in commercial operation
  - 8 PWRS in construction

- Evolutionary Pressurized Water Reactor for Next Generation Reactor
  - Based on Korean Standard Nuclear Power (KSNP) Plant
  - Two-loop design with two steam generators and four reactor coolant pumps
  - 4000 MW thermal power
  - Core Damage Frequency (CDF) < 1.0 x 10^{-5}
  - Extended design life of main components
  - Increased thermal margin
PURPOSE OF THIS STUDY

- Supporting Emergency Core Cooling System Design
  - DVI nozzle location
  - Safety injection flow rate

- Understanding Thermal-Hydraulic Phenomena Using Best Estimate Computer Code

- Comparing the Results from Conservative and Best Estimate Methods

- Checking the RELAP5 Capability for Large Break LOCA Application
ECCS CONFIGURATION

- EPRI ALWR Utility Requirements Document
- Four Mechanical Trains
  - Reduce core damage frequency
  - Simplify RCP discharge branch lines
- Direct Vessel Injection
  - No direct ECCW spillage to containment for cold leg break
  - Reduce the required capacity of ECCS
  - Increase the reliability of injection during LOCA events
- Four High Pressure Safety Injection Pumps Only
- Four Safety Injection Tanks
- In-Containment Refueling Water Storage Tank
- Two Emergency Diesel Generators
ANALYSIS METHODOLOGY

- Use of Conservative Initial and Boundary Conditions
  - Decay heat : $1.2 \times \text{ANS71}$
  - Peak linear heat generation rate : 450 W/cm
  - Axial power shape : peak at 65% elevation
  - 10% steam generator tube plugging
  - Single failure : one diesel generator failure
    (two trains are available)
  - Containment : assumption to minimize the containment pressure
    maximum net free volume
    minimum initial pressure
    maximum spray flow
COMPUTER CODES

- RELAP5/MOD3/K Merged with CONTEMPRT4/MOD5

- RELAP5/MOD3/K
  - Best estimate 1D two fluid code
  - Predict reactor coolant system thermal-hydraulic phenomena
  - Based on RELAP5/MOD3.1
  - Improvements to reflood model (NUREG/IA-0132, 1996)

- CONTEMPRT4/MOD5
  - Predict containment pressure
  - Provide back pressure for break discharge flow
<table>
<thead>
<tr>
<th>Location of Safety Injection Nozzle</th>
<th>No. of SI Pumps Available</th>
<th>No. of SI Pumps Injecting to Vessel or Cold Leg</th>
<th>Total Pump Flow Injected to Vessel or Cold Leg (kg/s)</th>
<th>No. of SITs Available</th>
</tr>
</thead>
<tbody>
<tr>
<td>Case A</td>
<td>DVI (Upper)</td>
<td>2 HPSI</td>
<td>120</td>
<td>4</td>
</tr>
<tr>
<td>Case B</td>
<td>DVI (Lower)</td>
<td>1 HPSI</td>
<td>120</td>
<td>4</td>
</tr>
<tr>
<td>Case C</td>
<td>CLI</td>
<td>2 HPSI</td>
<td>120</td>
<td>4</td>
</tr>
<tr>
<td>Case D</td>
<td>DVI (Lower)</td>
<td>1 HPSI</td>
<td>120</td>
<td>4</td>
</tr>
<tr>
<td>Case E</td>
<td>CLI</td>
<td>1 HPSI</td>
<td>120</td>
<td>4</td>
</tr>
</tbody>
</table>
DVI Nozzle Location
ANALYSIS RESULTS

• Base Case (Case A)
  - Rapid decrease of collapsed liquid level in downcomer and core after SIT empty
  - ECC bypass by entrainment and carry-over
  - No quenching for hot spot

• Elevation of DVI Nozzles
  - Comparison between cases A and B or between cases C and D
  - Elimination of carry-over for lower DVI nozzle

• Safety Injection Flow Rate
  - Comparison between cases A and C or between cases B and D
  - Increase of SI pump flow by replacing 2 HPSI with 2 LPSI pumps
  - Increase of SI flow is not efficient for upper DVI nozzle (case C)
  - Increase of SI flow is efficient for lower DVI nozzle (case D)

• Cold Leg Injection (Case E)
  - Combination of HPSI and LPSI pumps for each train
  - Total pump flow injected to cold leg is the same as that in cases C and D
  - Collapsed liquid level and fuel cladding temperature are in between cases C and D
Integration of Break Mass Flow
Collapsed Liquid Level in Downcomer
Collapsed Liquid Level in Core
Fuel Cladding Temperature at 3.05m from the Bottom of Active Core
EXPERIMENTAL RESULTS FROM 2D/3D PROGRAM

- Cylindrical Core Test Facility (CCTF)
  - Comparison between downcomer and cold leg injection
  - Periodic oscillation of pressures and fluid temperatures around downcomer was observed for downcomer injection
  - Water at the top of downcomer was subcooled in downcomer injection, whereas saturated in cold leg injection
  - Overall reflooding behaviour was almost the same in two cases

- Upper Plenum Test Facility (UPTF)
  - Flow condition in downcomer were highly heterogeneous
  - Water entrainment out the break and the reduction of downcomer water level were significant at high steam flows
  - Use of DVI can reduce effectiveness of ECC injection during blowdown (reduced penetration of water from accumulators to lower plenum)
  - Higher entrainment and level reduction at full-scale UPTF test than comparable subscale test
CONCLUSIONS

- Lowering DVI nozzle below the cold leg and increasing the injection flow after SIT empty will enhance ECCS performance for LBLOCA.

- RELAP5 seems to overpredict the ECC bypass under conservative initial and boundary conditions (limitation of 1D code application for multi-dimensional phenomena).

- Methodology using best estimate code with conservative initial and boundary conditions need to be verified further.

- The impact of reduced penetration for DVI shown from UPTF can be covered by Appendix K analysis.
RELAP5/PARCS
Generalized Thermal-Hydraulics/Neutronic Interface

Purdue University:
Prof. Tom Downar
Doug Barber

Nuclear Regulatory Commission:
Vince Mousseau
Dave Ebert

Presented By: J.M. Kelly

June 24, 1998
Coupled Thermal-Hydraulic / Neutronic Research

Purpose: Design a generic interface for coupling any Thermal-Hydraulics code with any Neutronics code.

- Preserve value of investment in interface.
- Make comparisons between different pairs of Neutronics and Thermal-Hydraulics codes easier.
- Do not require intimate knowledge of both the Neutronics code and Thermal-Hydraulics code to coupled them.

- Target codes:
  - Thermal-Hydraulics: TRAC-M, RELAP5
  - Neutronics: PARCS

Thermal-Hydraulic / Neutronic Interface Design
Thermal-Hydraulic / Neutronic Interface Design (cont.)

- Simple, well-defined Interface:
  - Performs mapping of property/solution data between Thermal-Hydraulic and Neutronic spatial domains.
  - Communicates calculational control information.

- Requires use of Code-Specific Data Map Routines.

- Minimal changes to the Thermal-Hydraulics and Neutronics codes.
  - Code-specific changes are localized.

- Thermal-Hydraulics code, Neutronics code, and General Interface are self-contained processes.

- Utilization of Parallel Virtual Machine (PVM) Software to handle communication.

Utilization of PVM In the Coupled Code
RELAP5 / PARCS: Communication Design

RELAP5-specific Data Map Routine (RDIMR)  General Interface (GI)  PARCS-specific Data Map Routine (PDIMR)

- recv GI ID
- send RELAP5 ID
- recv PARCS buffer
- send RELAP5 buffer
- recv 2 perm. matrices

- broadcast GI ID
- recv RELAP5 ID
- recv PARCS ID
- send RELAP5 buffer
- recv RELAP5 buffer
- send 2 perm. matrices

- send RELAP5 buffer
- send unpermuted RELAP5 vector
- recv RELAP5 buffer
- send permuted RELAP5 vector
- recv RELAP5 buffer
- send permuted RELAP5 vector

- recv PARCS buffer
- send permuted PARCS vector
- recv unpermuted PARCS vector
- send permuted PARCS vector
- recv unpermuted PARCS vector
- send permuted PARCS vector

RELAP5 / PARCS: Calculational Flow in RELAP5

Initialisation
- process initialization; communication setup; calculation setup

Input Processing
- yes

no

RELAP5-to-PARCS Mapping
- transfer initial condition data to PARCS

PARCS-to-RELAP5 Mapping
- obtain initial power from PARCS

Transient Iteration
- yes

no

Advance Thermal-Hydraulic Solution
- transfer time-dependent data to PARCS

RELAP5-to-PARCS Mapping
- yes

no

End
RELAP5 / PARCS: Calculational Flow in PARCS

Sample PWR Test Case: Problem Description

Purpose: Demonstrate the functionality of the coupled RELAP5/PARCS code.

RELAP5 Thermal-Hydraulics Model:
- Typical PWR problem: 4 loops with one of the loops undergoing a small break.
- Reactor trip on low pressure (1860 psi).
- Transient analyzed for 100.0 seconds.

PARCS Spatial Kinetics Model:
- Based on the NEACRP Benchmark core model (Full Core).
- Note: This model does not reduce to the point kinetics model used in the original RELAP5 Typical PWR input deck.
Sample PWR Test Case: Nodalization

Radial Nodalization:
(157 fuel assemblies, 64 radial reflectors)
- Thermal-Hydraulics: 1 channel
- Neutronics: 1 node / assembly (221 radial nodes)

Axial Nodalization:
- Thermal-Hydraulics: 6 volumes / 6 heat structure components for the single channel
- Neutronics: 18 axial nodes
- Axial domain boundaries are congruent (no overlap)
PWR Rod Ejection Analysis: Problem Description

Purpose: Demonstrate the fidelity of the coupled RELAP5/PARCS code.

- Control rod fully inserted at time 0.0 sec.; fully ejected at time 0.1 sec.
- Approximately 1.08$ of inserted reactivity.
- Plenum-to-plenum, quarter-core symmetric model is used.
  - Core is isolated from the system using infinite reservoirs at the inlet and outlet.
  - Time-dependent junction is used at the inlet to fix mass flow rate.
- Single phase flow calculation (no boiling occurs during the transient).
- Transient response is analyzed for 5.0 sec.

PWR Rod Ejection Analysis: Nodalization

Radial Nodalization:
(47 fuel assemblies, 17 radial reflectors)

- Thermal-Hydraulics: 48 parallel channels
  (no cross flow)
  - 1 channel / fuel assembly, 1 channel for all 17 radial reflectors
- Neutronics: 4 nodes / assembly

Axial Nodalization:

- Thermal-Hydraulics: 14 volumes / 14 heat structure components for each channel
- Neutronics: 18 axial nodes
- Axial domain boundaries are congruent (no overlap)
PWR Rod Ejection Analysis: Results

THREE DIMENSIONAL KINETICS COUPLING TO THERMAL HYDRAULICS

Joint effort of CEA/SERMA and CEA/SMTH

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- PURPOSE OF THE COUPLING
- GENERAL CONTEXT
- INFORMATION TRANSFER
- CODE MODIFICATIONS
- EXAMPLE OF COUPLING
- VALIDATION TESTS
- FUTURE WORK

OECD/CSNI Seminar on ‘Best Estimate Methods in Thermal-Hydraulics Safety Analysis’
Ankara, Turkey, 29 June-1July 1998
PURPOSE OF THE COUPLING

- Provide a tool for analysis of coupled multidimensional Kinetics and Thermal Hydraulics:
  - Rod ejection,
  - Instabilities,
  - Main Steam Line Break,...

- Minimal changes to each code.
  - Keep each code independent (maintenance, developments, ...)
  - Expertise focused (TH, kinetics)
GENERAL CONTEXT

Thermal hydraulics 6 equations model
CATHARE2 V14E

Thermal hydraulics 4 equations model
FLICA 4

Coupled TH-Kinetics
ISAS 1

3D Neutron kinetics
CRONOS 2.3

OECD/CSNI Seminar on 'Best Estimate Methods in Thermal-Hydraulics Safety Analysis'
Ankara, Turkey, 29 June-1 July 1998


**INFORMATION TRANSFER**

- The two codes are independent processes
- Communications are managed by ISAS
  
  *Integrated Safety Analysis System CEA/DRN/DMT/SERMA*
  
  ISAS1 based on P.V.M. → ISAS2 on CORBA

- ISAS allows to transfer:
  
  → data: a meshing + a set of variables,
  
  → commands: user defined commands.
For example:

CATHARE sends a vector of fuel center temperatures
+ a vector of fuel external temperatures,
ISAS builds a 3D field of fuel average temperature
and interpolates the field between the two meshes,
CRONOS receives a 3D field of fuel average temperature.
CODE MODIFICATIONS

- CATHARE:
  - Function to send/receive a command,
  - Function to send/receive any accessible TH variable (C.C.V.).

- CRONOS:
  - The developments already done for FLICA4-CRONOS coupling,
  - No specific developments.

- All modifications are located at the code interface: input data deck.
EXAMPLE OF COUPLING

CATHARE  
W fuel  
W fluid  

ISAS  

CRONOS  

pl, pv, o  
BORON C  
T center, UO2  
T external, UO2  

Nu20  
BORON C  
T average, UO2  

initialization  
Start code  
initialization  

if cmd = recv_power  
if cmd = send_TH  
if cmd = time_step  
if cmd = stop  
endif  

CRONOS send_power  
CATHARE recv_power  
CATHARE time_step  
CATHARE send_TH  
CRONOS recv_TH  
CRONOS time_step  

if cmd = send_power  
if cmd = recv_TH  
if cmd = time_step  
if cmd = stop  
endif
EXAMPLE OF COUPLING

• The previous example could be written:

  \[ \text{TH}^{n+1} = f(\text{kinetics}^n) \]

  \[ \text{kinetics}^{n+1} = f(\text{TH}^{n+1}) \]

• Other time schemes are possible, including parallel extension,

• Sub-cycling is possible for both codes (depending on the physics).

• Different methods will be tested,

• Best methods will be identified and maintained as referenced procedures.
VALIDATION TESTS

- $1 \times 1 \times 23$ Kinetics = $1 \times 1 \times 23$ TH done

  top deflector

  $1 \times 1 \times 23$ 3D kinetics

  bottom deflector

- Outlet boundary condition

  1D hydraulic channel
  × 23 axial meshes
  Fuel module

  Inlet boundary condition

- Boron concentration step at core inlet
- Liquid temperature step at core inlet

- $2 \times 3 \times 23 ; 3 \times 3 \times 23 ; 5 \times 5 \times 23$ in progress

- Comparison with SAPHYR system (CRONOS-FLICA)

OECD/CSNI Seminar on ‘Best Estimate Methods in Thermal-Hydraulics Safety Analysis’

Ankara, Turkey, 29 June-1 July 1998
FUTURE WORK

Planned:

- extension to CATHARE 3D module,
- VVER 1000 rod ejection transient (1/6 and 1/2 of the core modeled),
- RBMK rod ejection (10 fuel channels modeled separately),
- BWR stability study.

Foreseen:

- VVER 1000 Main steam line break,
- PWR Main steal line break,
- RBMK rod ejection (1480 fuel channels modeled separately).
THE NEED OF COUPLED 3D NEUTRONICS IN DBA AND BDBA ANALYSES USING CONSERVATIVE OR BEST-ESTIMATE APPROACH

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1. INTRODUCTION

Until recent years most of the thermal hydraulic safety analyses at least for PWRs have been made with codes which model the neutronics only with point kinetics. For BWRs, transient analyses have been carried out traditionally with axially one-dimensional models, because the coupling between neutronics and thermal hydraulics is very strong when there is boiling in the core, and the axial effects dominate. The stability considerations for BWRs always need the three-dimensional core models.

The need of coupled 3D neutronics calculation is largest in such cases where fission power development is important and its distribution changes during the transient. These cases include almost all Reactivity Initiated Accidents (RIA) especially if the reactivity increases due to complicated reasons, if the increase is asymmetric in the core, or if there are possibilities for recriticality in the later phase of the transient. All Anticipated Transients Without Scram (ATWS) need more complicated neutronics modelling than point kinetics because the fission power affects crucially the behaviour of the reactor during the whole transient and often very exotic flow conditions prevail in the primary circuit. During ATWS cases in beginning of cycle conditions there occurs heavy boiling also in the primary circuit of PWRs.

The licensing analyses for a nuclear reactor must be conservative, in the sense that the phenomena affecting the analyzed accident are exaggerated, and hence the safety margin of the analyzed case can be confirmed. The conservatism is needed because of limited knowledge but also because of the need to cover a great number of similar cases with one analysis, the bounding analysis. In nuclear power plants, thousands of parameters can affect the results of an analysis. Probabilistic methods have been tried in order to take into account the uncertainty in best-estimate transient results. However, only the variation of a few parameters can be handled without the number of analysis calculations becoming too large. In practice, good engineering judgement is the most important asset. In three-dimensional analyses the situation is even more complicated than in the models using point kinetics.

2. EXAMPLES OF ACCIDENT TYPES USEFUL TO ANALYZE WITH 3D MODELS

A typical example of a Design Basis Accident (DBA) which cannot easily be calculated without a three-dimensional core model is the Control Rod Ejection (CRE) accident, where considerable deformation of the radial and axial power distributions occur in the core. However, it has been usually analyzed with
point kinetics models by using conservative assumptions. The ensuring of the point kinetics analysis to be conservative is straightforward in CRE cases because only one goal is to be achieved: the maximization of the fission power developed during the transient. With optimization of fuel economy and tightening of safety requirements the need has risen to reduce the overconservatism also from these analyses.

The coupling between neutronic and thermal hydraulic modelling in the core is especially important in such CRE cases where the coolant mass flow in the initial state is lower than nominal, and there occurs boiling and even Departure from Nucleate Boiling (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 with 3D core models 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.

Steam line break is another type of design basis accident (DBA) 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. Although the total power level would not increase to the 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 (DNB). A three-dimensional model is also much more reliable in predicting the reactivity level after the trip.

Inhomogeneous boron dilution has recently been found out to be a possible cause of reactivity initiated accident in PWRs. Extensive studies on its consequences have been made e.g. in Finland. 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 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. 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. Such natural circulation conditions can be achieved e.g. during Small Break LOCA or ATWS cases.

In ATWS cases and other Beyond Design Basis Accidents (BDBA) the 3D core model can be used to eliminate any uncertainty due to spatial effects from the reactivity description. If the main circulation pumps stop during an ATWS the primary pressure increase and the amount of primary mass inventory lost depend crucially on the fission power behaviour of the core. The fission power distribution during the transition from flow rate of running main circulation pumps to two-phase natural circulation conditions is both axially and radially very different from the nominal conditions. The prevailing power level during this time period has a strong effect on the primary circuit thermohydraulics and thus on the primary mass inventory and the possible existence of the inherent boron dilution. Studies with different (1D or point kinetics) core models have indicated that reliable fission power prediction during this phase

582
of transient can be achieved only with a 3D core model describing neutronics and flow channels of the fuel assemblies individually. Using less dimensions in the core modelling did in some cases change the whole accident scenario of the plant e.g. with respect to the occurrence of inherent boron dilution, even though the model could separately be valid both in nominal and natural circulation flow conditions. Backfitting of the point kinetic constants to values derived from the 3D core model is at least needed. In most cases the axially 1D neutronics model calculates well the ATWS phenomena because the axial effect of density dominates again on natural circulation, when the radial effects between different channels have been equalized. The minimum requirement for ATWS incidents is the 1D neutronics model and the description of all circulation loops. The point kinetics model is not sufficient in any case. The 3D neutronics model is desirable.

Another kinds of BDBA scenarios are the transients and accidents happening in shut-down conditions. These types of accidents have a large impact on the PSA studies of the plants. In these exotic conditions all possibilities to increase the accuracy of neutronics calculation e.g. by using 3D models must be utilized. Often some recalculation of nuclear data used by 3D neutronics must be carried out.

3. MODIFICATION OF NEUTRONICS PARAMETERS

There is not very much experience in the world in carrying out conservative accident analyses with a best-estimate three-dimensional reactor dynamics code. The analyses reported are mostly made as best-estimate calculations where only the initial circumstances are chosen conservatively but the calculational parameters are not varied. Detailed instructions can only be given for distinct individual accident types, e.g. for a control rod ejection accident. In analyses of longer and more complicated accidents, e.g. ATWS, varying of key parameters can even change the nature of the accident.

Many neutronics parameters can or must be modified when conservative accident analyses are made with a three-dimensional best-estimate code. The most important of them are:

- reactivity feedback coefficients
- efficiency of control rods
- fraction of delayed neutrons
- fraction of decay power
- power distributions.

Different criteria are applied in the transient and accident analyses when the acceptability of the plant response to a disturbance is studied. For instance, maximum values of primary pressure, fuel enthalpy or cladding oxidation are limited. All these parameters do not always reach maximum values with the same combination of conservative modifications.

When using a three-dimensional core model the definition of conservatism is also not clear because modifications of different parameters cannot be made separately but they influence each other. A change of one parameter to conservative direction can decrease the supposed conservatism of some earlier change of another parameter.
The possible effects of all these modifications on the eigenvalue in steady state are compensated for by uniformly changing the fission neutron production cross-sections, which corresponds to the change in average burnup. The neutronics parameters must be changed concentratedly so that the conservatism of the calculations can be simply and reliably varied without changing the vast ordinary neutronics data.

3.1 Reactivity feedback coefficients

In all reactor dynamics analyses the conservatism of different modifications can change in different phases of the transient. It might be desirable to maximize the power level. During a power increase it is then conservative to decrease the absolute value of the reactivity coefficients of the negative feedback effects. However, during a power decrease, their values should be increased. Each transient must therefore be analyzed by performing sensitivity studies in order to know which phase of the transient is most critical.

The conservative values of fuel temperature reactivity coefficients can be obtained by uniformly adding to the fission neutron production cross-sections small linear contributions depending on the square root of the fuel temperature of each node. The moderator temperature reactivity coefficient is changed in the same way depending on the moderator density of each node. The reactivity coefficients will change according to the applied neutronics model when the conditions in the core change.

A more conservative value for the moderator reactivity feedback coefficient can also be achieved in a natural way by somewhat changing the boric acid concentrations from the critical value.

During ATWS situations, the alternative way to shut down the fission power is to increase the boron content of coolant using emergency cooling pumps. The efficiency of the increasing boron concentration to decrease the reactivity can also be tuned.

3.2 Efficiency of control rods

In principle the efficiency of control rods is modified for two reasons: the reactivity worth of them is not known exactly, or the analyses made in certain loading conditions are desired to represent more generally also other loadings where the reactivity worth of control rods can be slightly different. The modification can be made by changing the two-group constants or boundary properties which specify the control rods.

The reactivity worth of control rods is a key parameter in many types of analyses. Maximizing its value maximizes the power increase in different types of control rod withdrawal or ejection accidents. However, even the reactivity worth cannot be changed straightforwardly in the sense of conservatism: when maximizing the efficiency of the control rods also the effectiveness of the trip control rods is maximized and the core becomes more subcritical after the trip. In addition, a more effective control rod reduces the flux and power in its surroundings. Thereby the thermal hydraulics conditions in the initial stationary state are milder near the control rod - in the same area, where the consequences of the control rod movement would be strongest during the transient.

Usually the trip efficiency is minimized by assuming a control rod to be stuck in the upper position. The position of the stuck rod is chosen according to the maximum decrease of the trip reactivity worth or by choosing a control rod situated in the most critical area during each accident.
Maximizing the initial insertion of the control rods is a natural way to make conservative calculations.

3.3 Fraction of delayed neutrons

The fraction of delayed neutrons is one of the decisive parameters which change during the fuel cycle. It is larger at the beginning of the cycle than at the end, and it typically has an effect on the time at which the analyses of some accidents are most critical during the fuel cycle. The uncertainty of its value is often compensated for by using a somewhat decreased value, when analyzing fast power bursts.

3.4 Fraction of decay power

The conservatism of maximized fraction of the decay power is clear in the analyses of loss of coolant type accidents. On the contrary, in reactivity initiated accidents it is conservative to assume a minimized fraction of decay power so that the prompt power peak is maximized. However, if transients beginning with power increase are calculated as ATWS, where the power decrease is not induced by the trip, the situation is no more clear. Then the different acceptability criteria may demand different directions for the conservative changes of the decay power fractions.

3.5 Power distributions

In 3D analyses even the power distribution can be conservatively varied. The best-estimate radial power distribution is usually not changed but the axial power distribution is often modified. The behaviour of the hottest fuel rod and the hottest flow channel are usually analyzed in separate calculations with the axially one-dimensional code based on the fission power behaviour results of the 3D calculation. The conservatism for the assembly peaking factor can be included in the fuel rod peaking factor as an extra multiplier. The engineering (safety) factor is usually also included as a multiplier. The given axial power distribution of a fuel assembly is multiplied with this combined hot channel factor.

In the hot channel analyses the axial power distribution cannot usually have its best-estimate form because the maximum local linear power is often given as a limiting value. However, already the experience from the analyses made with the axially one-dimensional dynamics codes has shown that the results can be seriously distorted if the axial power distribution is changed only in the hot channel analyses. The neutronics is not calculated in the separate hot channel analyses. In the case where the axial distribution is changed only in the hot channel, the strong axial reactivity feedback effects of coolant have been affecting a very different axial distribution. Therefore, if it is necessary to change the initial axial power distribution, it should be modified already in the 3D calculation itself by changing the mutual reactivities of the nodes at different radial levels.

With the modification of the axial power distribution the maximum local linear power of the fuel rod can be changed. The modification also affects the reactivity worth of the control rods inserted partially into the core. In addition it affects the tendency to achieve the departure from nucleate boiling conditions, which most probably occurs in the upper part of the core with larger void fraction.

Physically the changing of axial power distribution can be thought to represent a limiting xenon oscillation phase.
4. HOT CHANNEL METHODOLOGY IN CONNECTION WITH 3D CORE MODELLING

When using point kinetics or axially one-dimensional calculations there is only one possibility to calculate the hot channel. In three-dimensional analyses, there are as many different possibilities for hot channel calculations as there are fuel assemblies, but the practical work to make the calculations and to handle their results brings restrictions. If only one very conservative hot channel represents the whole core, the benefit of the three-dimensional calculation is lost.

If the hot channel analyses are included directly in the three-dimensional analysis, very large computer capability is needed. There are also difficulties to parametrically handle conservatisms of the hot channels and every different loading must be analyzed individually.

In Finland, a new multiple hot channel methodology is used in connection with 3D core modelling, which is in each calculated case based on the analysis of the power behaviour in different parts of the core.

The main steps of the methodology are:
- The hot channel analyses are made separately using data files containing axial power distributions and pressure differences over the core.
- The core assemblies are grouped together to a few areas according to time histories of assemblies during the transient.
- Every group is represented by one hot channel.
- Hot channel factors in every group have their maximum credible values.
- The results of a typical loading can be used for all similar cycles.
- Applying of conservatisms is easy.

The core is divided into areas with differing time histories. A control rod ejection accident example can be seen in Fig. 1., where the radial fission power distributions and their relative changes can be seen at different phases of the transient. In this case, the magnitude of the disturbance and thus the grouping of the fuel assemblies depends on the distance of the assemblies from the ejected control rod. Each area is studied with one representative core location. At each location a constant excess radial peaking factor is applied to generate a hot fuel rod. The maximum value for this excess factor is determined so that it produces the maximum permissible rod peaking either in the actual initial state or in the reference full power state, whichever is more limiting. Rather than a real rod of the specific core, this represents a potential hot fuel rod at the given location. Consequences for fuel rods of lower power are then studied by reducing the excess peaking factor and repeating the hot channel simulation over time. Also the RIA criteria limits for the departure from nucleate boiling (DNB) or oxidation of the cladding material are found with this methodology.

Comprehensive sensitivity studies can easily be done with different fuel and channel properties or different correlations in order to check the conservatism of the results. These hot channel calculations are very fast with the axially one-dimensional code. The properties of the gas gap between the fuel and cladding is a typical example of important data which can not be known exactly. In RIA cases the maximum enthalpy of fuel pellet can be confirmed only with sensitivity calculations because on one hand the temperature dependent over-estimate values for the gas gap conductace enhance the appearance of DNB which increases the fuel temperature, and on the other hand, this kind of gas gap reduces the initial fuel temperatures.
Radial Fission Power Distribution
in Stationary State

Radial Fission Power Distribution
at Time 1.6 s (after Ejection)

Ratios of Assembly Fission Powers
at Time 1.6 s to Stationary Powers

Ratios of Assembly Fission Powers
at Time 16 s to Stationary Powers

Figure 1. Control rod ejection accident example. Radial fission power distributions and relative changes of them.
5. FINNISH 3D REACTOR DYNAMICS CODES

In Finland a number of transient and accident analysis codes have been developed during the past twenty years mainly for the needs of our own power plants, but some of the developed methods have also been utilized elsewhere. In these codes the strong coupling between neutronics and thermal hydraulics inside the reactor core has been solved by using iterative simultaneous solutions of equations describing these different physical phenomena. The coupling is especially strong when boiling is occurring in the core. Two different solutions have been used to perform the coupled calculations with cooling circuits. In BWR models the coolant circuit has been included as essential part together with the core flow equation solution. In PWRs (or VVERs) the circuits have been calculated with a separate model connecting with the core model only by data changing once during a time-step.

TRAB-3D and HEXTRAN are the Finnish 3D reactor dynamics codes for rectangular (BWRs and PWRs) and hexagonal (VVERs) fuel lattices, respectively. The rectangular TRAB-3D is a new code, therefore the applications for BWRs have been made so far with the axially one-dimensional TRAB code. TRAB-3D validation is continuing against measurements of the Finnish BWRs. The PWR coupling of TRAB-3D will be tested in the OECD/NEA/NSC Main Steam Line Break benchmark problem. Examples of HEXTRAN applications can be found in References 4 - 6. The typical accidents analyzed for VVERs are the same as for the other types of PWRs.

REFERENCES


Thermal-Hydraulic Challenges in Advanced Reactor Designs

Presented to OECD/CSNI Seminar on "Best Estimate Methods in Thermal-Hydraulic Safety Analysis"

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Office of Nuclear Regulatory Research

July 1, 1998

Thermal-Hydraulic Challenges in Advanced Reactor Designs

 CONTENTS:

◆ Overview of NRC/RES AP600 Program
◆ AP600 System & SBLOCA Description
◆ Modeling Difficulties & Resolution
◆ Numerical Considerations
◆ Summary

July 1, 1998

589
NRC/RES AP600 PROGRAM

■ OBJECTIVE:
  - Provide T/H analysis tools to NRC/NRR that have been demonstrated to be applicable for the AP600 design.

■ APPROACH:
  - Use TRAC-P for LBLOCA Analysis
  - Use RELAP5 for SBLOCA and Operating Transients

  - Major effort was expended on the applicability program - code assessment & improvement - for SBLOCA and operational transients.

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NRC/RES AP600 PROGRAM

■ Focus on SBLOCA & Transients?
  - LBLOCA phenomena do not appear appreciably different than for current generation PWRs.
  - Potential for interaction between passive systems, or between non-safety active systems and passive systems.
  - Potential for non-limiting transients (eg, SGTR) to activate ADS thereby becoming a LOCA.
  - Passive systems drive flows with small heads, requiring more precision in analysis.
    - Same confidence level in analysis?
NRC/RES AP600 PROGRAM

- PIRT Based Code Assessment
  - Integral Test Assessment:
    - 3 Facilities with Different Scales
      - SPES, OSU & ROSA
    - Focus on 4 Transients:
      - 1" Cold Leg Break
      - 2" Pressure Balance Line Break
      - 200% DVI Line Break
      - Inadvertent ADS Actuation
  - Separate Effects Test Assessment:
    - High Ranking PIRT Phenomena

NRC/RES AP600 PROGRAM

- RESULTS: Integral Test Assessment
  - For each transient, facility & PIRT phenomena:
    - Code assessment judgments were made:
      - Excellent, Reasonable, Minimal, Insufficient
  - PIRT High Ranked Phenomena:
    - NO code models judged insufficient
    - Excellent: 6%
    - Reasonable: 85%
    - Minimal: 9%
  - RELAP5 (pre-release version of Mod 3.3) judged applicable for AP600 analyses.
AP600 System & SBLOCA Description

- **Passive Safety Systems:**
  - Core Make-up Tanks (CMT):
    - 2 high pressure tanks using gravitational potential to provide borated water to primary system (HPSIS).
  - Accumulators:
    - 2 gas pressurized accumulators providing water at high flow rate in event of LBLOCA.
  - Automatic Depressurization System (ADS):
    - 2 Sets of Stages 1-3 valves blowdown from top of pressurizer through sparger into IRWST.
    - 2 Sets of Stage 4 valves discharge from hot legs to containment atmosphere.

AP600 System & SBLOCA Description

- **Passive Safety Systems:**
  - In-Containment Refueling Water Storage Tank (IRWST):
    - Large tank to provide gravity fed cooling water for long-term cooling.
  - Passive Residual Heat Removal System (PRHR):
    - HX submerged in IRWST that provides natural circulation loop (hot leg to steam generator inlet) for passive RHR.
  - Direct Vessel Injection (DVI):
    - Inject ECCS water (CMTs, accumulators & IRWST) at a level below cold legs to minimize ECCS bypass.
AP600 System & SBLOCA Description

- Pressure history for SBLOCA:

![Pressure History Graph](image)

Modeling Difficulties & Resolution

- AP600 SBLOCA Analysis Difficulties:
  - Primarily due to:
    - Low pressure (esp. boiling & condensation)
    - Buoyancy driven flows (small frictional losses)
    - Presence of noncondensible gases in primary system
  - Similar to problems encountered in the analysis of transients such as mid-loop operation with loss of RHR in standard plants.
Modeling Difficulties & Resolution

Topics:

- Code failures due to non-condensible gases

- Critical Flow:
  - Artificial choking at low pressure
  - Thermal non-equilibrium effects

- Vessel (Core) Inventory:
  - Effect of recirculating flows
  - Void oscillations
  - Void fraction at low pressure

- Progression from inability to perform calculation to a concern as to the degree of accuracy.

Modeling Difficulties & Resolution

- Critical Flow Model
  
  - Symptom:
    - ADS-4 mass flow rate under-predicted by up to a factor of 10 for low pressure, low quality two-phase flow.

  - Root Cause:
    - Standard RELAP5 model has a flaw in choking criterion when phasic relative velocity is significant.

  - Effect:
    - System depressurization was impeded, leading to intermittent IRWST injection.

  - Resolution:
    - Implemented Henry-Fauske model as option.

July 1, 1999
Modeling Difficulties & Resolution

- Critical Flow
  - Effect of artificial choking of ADS-4 for low pressure two-phase flow on IRWST injection:

![Graph showing data and RELAPS predictions over time.]

Modeling Difficulties & Resolution

- Critical Flow
  - Thermal non-equilibrium effects on subcooled break flow for thin orifice plate:

![Graphs showing data and RELAPS predictions for critical flow over time.]

July 1, 1998
Modeling Difficulties & Resolution

- 'Quasi-3D' Model for OSU:

Modeling Difficulties & Resolution

- Core Collapsed Liquid Level
  - Effect of unphysical two-phase recirculating flow in the core for "quasi-3D" model of OSU:
Modeling Difficulties & Resolution

- Core Collapsed Liquid Level
  - Use of 1-D noding for core greatly improves prediction:

- Core Collapsed Liquid Level
  - Close-up of ADS-4 blowdown to IRWST injection shows under-prediction of inventory & void oscillations:
Modeling Difficulties & Resolution

- Core Collapsed Liquid Level
  - Momentum flux work around for downcomer & code improvements => better prediction of inventory & smaller void oscillations.

![Graph of Core Collapsed Liquid Level](image1)

Modeling Difficulties & Resolution

- Void Fraction Oscillations
  - Core vapor generation rate during ADS-4 boil-off period exhibits oscillations rather than decay curve.

![Graph of Void Fraction Oscillations](image2)
Modeling Difficulties & Resolution

- Void Fraction Oscillations
  - Map of liquid interfacial heat transfer: \( h \cdot \Delta T \)

- Void Fraction Oscillations
  - Code improvements \( \Rightarrow \) core vapor generation rate during ADS-4 boil-off period is less oscillatory.
Modeling Difficulties & Resolution

■ Core Void Fraction - Accuracy
  ● Interfacial drag model over-predicts shear for low pressure decay heat conditions => core level ~10% low.

![PERICLES End-of-Refuel](chart)

Numerical Considerations

■ Further Improvements Needed:
  ● Computational Efficiency
    ▪ More implicit solution techniques
    ▪ Minimize 'numerical events' that reduce time step
    ▪ Investigate use of parallel processing
  ● Code Robustness
    ▪ Improve ability of code to run transient to completion without requiring user intervention
    ▪ Involves: phase appearance/disappearance, water packing, implicit wall heat transfer, time step control...
  ● Code Accuracy (Numerical)
    ▪ Higher order differencing techniques (minimize diffusion)
    ▪ Better mass conservation for long transients
Summary

- RELAP5 Applicability to AP600:
  - Code improvements & work-arounds were needed.
  - Mod 3.3 (pre-release) was judged to be applicable:
    - NO models for high ranking PIRT phenomena were judged to be insufficient, 91% judged reasonable or excellent.

- Code Model Improvements Desired:
  - Thermal stratification in pools & horizontal pipes.
  - Interfacial friction for low pressure.
  - Tee Offtake Model.

- Future Improvements Needed:
  - Efficiency, Robustness & Accuracy.
BEYOND DESIGN-BASED ACCIDENT (BDBA)

PHENOMENA AND RISK REDUCTION ISSUES

Dr. Themis P. Speis
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Beyond Design-Based Accident (BDBA)

Phenomena and Risk Reduction Issues

Dr. Themis P. Speis
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Abstract

Today most, if not all, OECD member countries with commercial nuclear power programs consider in the licensing/regulatory safety reviews events (internal and/or external to the plant) which could lead to substantial core damage of the reactor core, whether or not there are serious off site consequences. This was not the case early-on when most of the world's operating nuclear power plants (NPPs) were designed and constructed. Then those plants were designed using as a basis an envelope of "design basis events" which were developed over the years and used to test the overall adequacy of the NPP design. These "design basis events" were believed to represent a sound composite engineering judgment regarding the reasonable upper boundary for events which might occur. Also, they were thought to define a reasonable envelope of all credible events. Thus plants were required to be designed to mitigate the consequences of those events considered credible. The most severe in this set of "design basis events" in terms of challenging both the reactor and its associated systems and the containment is the large break Loss-of-Coolant (LOCA) accident. The characteristics of this accident serve to set the requirements for a number of safety systems (e.g., ECCS), including the design of the containment building. Events which could lead to a severe accident were considered incredible, and thus not considered in the design. This approach to licensing/regulation is also referred to as "Deterministic-Based" as opposed to "Risk-Based," even though the deterministic approach contains implied elements of probability. For example, reactor vessel rupture is considered too
improbable to be included as an accident to be analyzed; also the likelihood that a single emergency core cooling system or system train would not function was considered high enough that safety train redundancy and protection against single failure were required.

Operational experience, especially the accidents at TMI-2 and Chernobyl, and the knowledge and insights gained from the many Probabilistic Risk Assessments (PRA) have changed all that and led us to our current understanding of event sequences as a continuum of probabilities of occurrence, and has led to a modification of safety review requirements and the consideration of a spectrum of failure sequences well beyond that of the traditional design basis approach. The above mentioned accidents as well as the results from the PRAs performed have also indicated that the conditions that could develop in such severe accidents as a result of the thermal-hydraulic material interactions which take place could be more severe than those chosen for "design basis accidents," and that failure of the containment, in some instances early into the accident phase, was a possible outcome of such accidents

This paper discusses how the licensing/regulatory review has evolved, especially after the TMI-2 and Chernobyl accidents and how the severe accident issue has been addressed for operating nuclear power plants and what has been the role of PRA and severe accident research in the selection of severe accidents from the spectrum of severe accidents considered, and some of the consequential backfits made. The paper also addresses the use of conservative vs. best-estimate methods used in severe accident analysis. Finally, the paper will discuss how severe accident sequences are being addressed in new nuclear power plant designs, whether they are of an evolutionary type (improvement to existing designs) or a passive one (more radical in some aspects of their design against accidents including those that lead to core melt)
provide a list of the most important severe accident related loading to containment(s) based on the many PRAs and severe accident research performed mostly in OECD/NEA member countries [see: "Nuclear Safety Research in OECD Countries," report by an NEA group of experts, OECD 1994].
OUTLINE

- Evolution of licensing/regulatory reviews
- Severe accident considerations
- Reactivity vs. core uncoverey actions
- A graphic presentation of containment loadings in LWR severe accidents
- A quantitative example of a severe accident
- PRA experience
- Approaches(s) in addressing severe accidents for operating plants
- Approach(es) in addressing severe accidents for future plants
- Containment performance criteria for future plants
- List of severe accident phenomena and associated containment loadings
LICENSING/REGULATORY REQUIREMENTS/APPROACH

- DEFENSE-IN-DEPTH (MULTIPLE, SUCCESSIVE BARRIERS)
  - Design for Normal Operation (Emphasize Equipment Reliability, Redundancy and Inspectability)
  - Design to Detect Failure(s) and Shut Plant Down
  - Design to Control the Consequences of More Damaging Accidents

- DESIGN BASIS EVENTS
  - Transients (AOOs)
  - Accidents

- CONTAINMENT [AND OTHER SAFETY SYSTEM(S)] DESIGN
  - DBAs (e.g., LOCA, SLB)
  - External Events
  - TID-14844 Fission Product Source Term
EVOLUTION OF LICENSING/REGULATORY REQUIREMENTS/APPROACH
(Continued)

- TMI experience feedback
  - Multi-failure Considerations (Plant Equipment Systems and Operators)
  - Symptom-Oriented Emergency Operating Procedures
  - Design for Hydrogen Release from Core Degraded Accident for “Weaker” Containments

- Continuing Operating Experience and PRA Insights Feedback
  - Revised and/or New “Regulations” (e.g., ATWS, Station Blackout)
  - Evaluation of Operating Experience for Precursor Events
  - Performance Indicators

- Severe Accident Considerations (Accidents More Severe than DBAs)
SEVERE ACCIDENTS*

"SEVERE ACCIDENTS ARE THOSE EVENTS WHICH ARE BEYOND THE SUBSTANTIAL
COVERAGE OF DESIGN BASIS EVENTS AND INCLUDE THOSE FOR WHICH THERE IS
SUBSTANTIAL DAMAGE TO THE REACTOR CORE WHETHER OR NOT THERE ARE
SERIOUS OFFSITE CONSEQUENCES."

First examined systematically in WASH-1400 using Probabilistic Risk Assessment techniques.

* US NRC's Definition
SEVERE ACCIDENT CONSIDERATIONS

- WASH-1400, Other PRAs, TMI-2 and Chernobyl Accidents, All Tell Us That Severe Accidents Represent the Major Contribution to Risk from Commercial Nuclear Power Plants.

- Margins (To Severe Accident Challenges) in Existing Plants;

- Importance of Containment;

- "Practical Improvements" to Existing Plants;


- Approaches Issues to Addressing Severe Accident Phenomena in Future NPP Designs

- Defense-in-Depth into the Severe Accident Domain (i.e., Containment performance given severe accident)
- WASH-1400
- TMI
- PRAs/Research
- Chernobyl

Severe Accidents in Licensing/Regulation
- Re-Assess Risk
- Prevention vs Mitigation (balance?)
- Emergency (S.A.) Procedures
- Risk Reduction Enhancements to Containment Performance
REACTIVITY VS CORE UNCOVERY ACCIDENTS

Chernobyl

3,000 P₀ (?)

Power

ms

t

TMI

System Pressure

Coolant Pumps Off

Block Valve Closed

Initial Core Heat Up

Degraded Core Heat Up

Core Relocation

Time (minutes)

Fuel

Water

S.E.?
CONTAINMENT LOADS

- \((H_2 + CO) + O_2 \rightarrow P\)
- Melt + Air \(\rightarrow P\)

Energetic vs. Quasi-Static Events

Coolable Debris Bed

Penetration
INITIAL CONDITIONS
P = 100 kPa ≈ 1 ATM
T = 298 K = 25C
AIR SATURATED WITH WATER VAPOR

FIGURE 7
PRESSURE AND TEMPERATURE AFTER HYDROGEN-AIR COMBUSTION. CONSTANT VOLUME AND ADIABATIC
FIGURE 2-3 Reactor building pressure versus time at TMI-2.

Source: EPRI (1980).
HUMAN ERRORS, RECOVERY ACTIONS AND PROCEDURES, TEST AND

MAINTENANCE

SUPPORT SYSTEM VULNERABILITIES

DOMINANT ACCIDENT INITIATORS

GENERIC AND PLANT-SPECIFIC CONTAINMENT VULNERABILITIES

RISK CONTRIBUTING UNCERTAINTIES

MOST RISK CONTRIBUTORS ARE PLANT SPECIFIC
APPROACH(ES) TO ADDRESSING SEVERE ACCIDENTS FOR OPERATING PLANTS

- Systematic examination of individual plants to identify plant-specific vulnerabilities to severe accidents (IPE Program in the USA) - plant specific and/or generic PRAs in many OECD member countries

- Development of Accident Management Procedures (Utilization of Insights from plant-specific/generic PRAs and Severe Accident Research)

- Various Methods/Approaches to reducing the probability of containment failure, e.g.,
  - Accident Management Procedures (availability of water on the drywall floor of Mark Is to avoid early failure of Mark I liner)
  - Hydrogen control measures (containment inerting and/or use of hydrogen igniters)
  - Containment Depressurization
    - Installation of FVCS in all French PWRs
    - Venting of Mark Is via suppression pool in the USA (serves both core melt prevention and containment failure mitigation)

- Variety of regulatory decision criteria
  - Risk analysis including cost-benefit considerations
APPROACH(ES) TO ADDRESSING SEVERE ACCIDENTS FOR OPERATING PLANTS
(continued)

- Availability of containment margins to severe accident challenges
- Relative importance of severe accident containment failure modes (Early vs. late failures)
- No radioactivity releases from "credible" severe accident containment failure modes (the French approach)
- Best-estimate methods and proper consideration of uncertainty
## Failure Modes in Mark I Containments

<table>
<thead>
<tr>
<th>Failure Mode</th>
<th>Risk Importance</th>
</tr>
</thead>
<tbody>
<tr>
<td>1. Overpressurization: Overpressurization leading to core damage (i.e., containment failure before core melting)</td>
<td>YES+</td>
</tr>
<tr>
<td>2. Steam explosion: Missile</td>
<td>NO</td>
</tr>
<tr>
<td>3. Failure to isolate*</td>
<td>NO</td>
</tr>
<tr>
<td>4. Hydrogen burn/detonation</td>
<td>NO</td>
</tr>
<tr>
<td>5. Overpressurization: (Corium/concrete interaction plus steam)</td>
<td>YES</td>
</tr>
<tr>
<td>6. Overtemperature: (Corium/concrete interaction)</td>
<td>YES</td>
</tr>
<tr>
<td>Failure Mode</td>
<td>Risk Importance</td>
</tr>
<tr>
<td>--------------</td>
<td>-----------------</td>
</tr>
<tr>
<td>7. BASEMAT MELT-THROUGH: (CORIUM/CONCRETE INTERACTION)</td>
<td>NO</td>
</tr>
<tr>
<td>8. CONTAINMENT SHELL (STEEL LINER) MELT-THROUGH</td>
<td>VARIABLE**</td>
</tr>
<tr>
<td>9. INTERFACING LOCA: (CONTAINMENT BYPASS)*</td>
<td>NO</td>
</tr>
</tbody>
</table>

* Mitigation features are ineffective against these failures. Their probability can be reduced by procedural/design changes.

** Depends on vessel failure mode, corium's ability to flow to and melt through the liner, especially in the presence of water.

+ In the absence of wetwell venting.
APPROACH(ES) TO ADDRESSING SEVERE ACCIDENTS FOR FUTURE PLANTS

- Variety of approaches being considered, e.g.,:
  - codify certain design features which reduce the probability and/or accommodate the consequences of severe accident challenges to the plant
  - codify the severe accident phenomena which a designer must consider in the design
  - basic thrust of above is the consideration/allowance of severe accidents in the design

- Utilization of the basic DBA approach with its primary focus on avoiding a severe accident, but AUGMENTING it with severe accident considerations

- As for operating plants, a key issue involves the use of PRA in decision-making, including the state of knowledge and understanding of the severe accident phenomenological issues which are crucial in the quantification of Containment Event Trees (CETs), e.g., availability of containment margins to severe accident challenges, even if core-damage frequency is very low (10^{-6} and below); is there still a need for defense-in-depth in the severe accident domain?
CONTAINMENT PERFORMANCE CRITERIA FOR FUTURE PLANTS

DBA Approach Augmented by Severe Accident Considerations

i.e., start with "current" design basis approach and look for "enhancements" to containment performance to attain margins to severe accident challenges (to sequence beyond the design basis)

Utilize extensive information from PRAs and severe accident research (analytical and experimental) to:

- identify risk dominant vulnerabilities or challenges to containment

- develop deterministic criteria to address each one of these vulnerabilities or challenges, e.g., threat from hydrogen, corium-concrete interactions, corium-containment boundary interaction, etc.

- e.g., to reduce the likelihood of containment failure from \( \mathrm{H}_2 \) buildup, the containment systems shall have sufficient margin in the design such that, in the event of accidents that lead to significant core damage, containment integrity will be maintained assuming a reaction of steam with 100% of the Zr in the active core region.

   This could be accomplished via

   A. providing an inerted containment (\(<4\% \, \mathrm{O}_2\)) or

   B. providing a containment design which precludes global concentrations of \( \mathrm{H}_2 \) from exceeding 10% and can maintain its integrity considering the effects of the burning and local detonations.
CONTAINMENT LOADINGS IN SEVERE ACCIDENTS

The severe accident phenomena that are capable of generating a much higher level of loading than that considered in the design basis may be summarized as follows in terms of nomenclature employed in WASH-1400 for PWRs. In general, the key phenomena are generic in nature but can affect containment performance in various ways depending on reactor type, containment size and configuration and/or other unique design features associated with a specific system.

*Alpha* ($\alpha$): Large scale in-vessel molten core-water energetic interaction, usually referred to as steam explosion, with potential to cause early containment failure via energetic missile penetration of the containment.

*Gamma* ($\gamma$): Hydrogen deflagration or detonation, it may or less important according to the type of reactor and containment type.

*Early Delta* ($\delta_e$): Rapid containment overpressurization early in the accident.

*Delayed Delta* ($\delta_d$): Gradual overpressurization at a late stage of the accident.
CONTAINMENT LOADINGS IN SEVERE ACCIDENTS
(CONTINUED)

Epsilon (ε): Basemat melt-through via its decomposi-
tion from contact with high-temperature core
debris.

Beta (β): Failure to isolate the containment.

V : Containment bypass (e.g., interfacing system LOCA).

Other severe accident phenomena that were not
addressed in WASH-1400, but were recognized to
challenge containment integrity in the recent past
include direct containment boiling (DCH) and BWR
Mark I liner failure:

DCH : Involves melt release at high pressure and
the potential for large-scale melt dispersal
throughout the containment volume. Such
dispersal could lead to direct boiling of the
containment atmosphere and associated
postpurification. Similar to the Early Delta.

Mark I failure : Potential direct exposure of the liner
in the vicinity of the drywell floor
with high-temperature melt can provide
an obvious mechanism for liner failure
(melt-through or creep failure). A variant
of the Epsilon.
Seminar on Best Estimate Methods in Thermal-Hydraulic Safety Analysis
of OECD/NEA

Steering Committee for Nuclear Energy/Committee on Safety of Nuclear Installations
Principal Working Group No. 2 on Coolant System Behaviour

held in Ankara, Turkey, June 29th to July 1st, 1998
hosted by Turkish Atomic Energy Authority

A New Safety Approach for Future PWRs
M. Champ, IPSN and R. Kirmse, GRS
presented in Session 4 (Selected Issues)

ABSTRACT

The subject of this paper is the development of a new safety approach for future PWRs. It describes both the procedure of the process and the technical content of the common recommendations achieved until now.

The approach is being developed according to the following procedure: the technical consultants GRS (Gesellschaft für Anlagen- und Reaktorsicherheit mbH) and IPSN (Institut de Protection et de Séreté Nucléaire) elaborate the technical basis. Starting from essential background information, common positions of GRS and IPSN are worked out in the form of comments, recommendations for technical positions, and identification of the need for further information. This work is the basis for a treatment of the subjects within the technical advisory bodies to the safety authorities GPR (Groupe Permanent chargé des Réacteurs Nucléaires) and RSK (Reaktorsicherheitskommission) in joint meetings. These activities result in GPR/RSK recommendations, which are submitted to DFD (German-French Directorate) for adoption.

The first set of recommendations has been issued in 1993. It contains among others the following safety objectives:

- A further reduction of the core melt frequency.
- The „practical elimination“ of accident situations which could lead to large early releases of radioactive material.
- For low pressure core melt situations the design has to be such that the associated maximum conceivable releases would necessitate only very limited protective measures in area and time (no permanent relocation, no need for emergency evacuation outside the immediate vicinity of the plant, limited sheltering, no long-term restrictions in the consumption of food).
Since 1994, the common safety approach has been further developed, extended and refined, especially in the following areas:

- Severe Accidents (general approach, radiological consequences, impact on containment design, R&D needs)
- System Design and Use of PSA (general approach, use of PSA, primary system overpressure protection and depressurisation, reliability of shutdown function, consideration of shutdown states, secondary side heat removal, electrical power supplies, residual heat removal, secondary side overpressure protection, etc.)
- Internal and External Hazards
- Primary Circuit Integrity (break preclusion concept)

In the presentations some examples of achievements will be given for subjects recently treated (severe accidents, system design), and the common recommendations will be presented and explained.

1 Introduction

The co-operation between the French and German safety authorities has been initiated already in the early seventies by establishing the DFK (Deutsch-Französische Kommission). The activity of this commission was focused on issues arising from nuclear power plants situated in France and Germany close to the common border and included a safety comparison between French and German plants. In 1990 the cooperation was significantly extended by the establishment of the DFD (Deutsch-Französischer Direktionsausschuss) of BMU (Bundesministerium für Naturschutz, Umwelt und Reaktorsicherheit) and DSIN (Direction de la Sûreté des Installations Nucléaires). One of the aims of DFD is the harmonisation of the regulatory approach in both countries with special emphasis on the development of common positions and safety principles for the design of future pressurised water reactors.

The development of national nuclear safety standards in France as well as in Germany was closely linked to the national nuclear programme and to the national technological environment. For this reason alone, it is not surprising that safety requirements are not identical in both countries. One main objective is to develop a common French-German safety approach despite of the differences in licensing rules and practices for existing plants. This process is rather straightforward in new areas without existing regulation (e.g. mitigation of severe accidents) and more complicated in areas with existing but different licensing practices (e.g. air plane crash).
Important features of the content of the French-German safety approach for future PWRs are outlined in the paper. Basically, the new safety approach was drafted by the technical safety organisations on request of the French and German safety authorities in close co-operation with their advisory bodies.

The development is a continuing process, which started with the development of basic safety objectives (issued in May 1993), followed by the treatment of key safety issues (1993-1994) and a further refinement of the key issues and the treatment of other important safety issues (since 1995, still continuing).

This stepwise procedure allowed the industry to take into account the content of the approach during their design development. The designer's Conceptual Safety Features Review File (CSFRF) was issued in September 1993 after the basic safety objectives had been issued in May 1993. For the Basic Design Report of the EPR, delivered to the safety authorities in the autumn of 1997, the designer was able to take into account the content of the safety approach for the key issues (beginning of 1995) and a large part of the subsequent refinement phase.

During the refinement process of the development of the common safety approach the design information of the CSFRF and later documents provided by the designer on the design progress have been taken into account. This iteration process was advantageous among others for two reasons: firstly, the area and extent of necessary refinement was easier to estimate on the basis of a design concept and, secondly, potential inconsistencies between the basic safety objectives and first design approaches have been identified in an early phase of the development.

2 Development and Content of the Common Safety Approach

This chapter presents the development of the common safety approach from the viewpoint of the technical safety organisations, concentrating on the technical aspects and the safety strategy rather than on particular legal aspects, which are in the responsibility of the governments.

The first set of common recommendations which represented the general safety approach was issued in May 1993 as "GPR/RSK Proposal for a Common Safety Ap-
proach for Future Pressurized Water Reactors". It was adopted by DFD in June 1993. This document, which contains general safety objectives and technical safety principles, is hereafter called the "Common Safety Approach of 1993". It is the basis for all further more detailed and refined recommendations and requirements.

The content of this basic approach, developed by IPSN/GRS, recommended by GPR/RSK and adopted by DFD is summarised briefly. It contains general safety objectives and technical principles for their realisation. They are considered to be an important basis for the future work.

The common opinion of the groups of experts is that the significant improvement aimed at for the next generation of PWRs (to be constructed at the very beginning of the next century) can be obtained in the "evolutionary" way if due consideration is given to the lessons learned from operating experience and from in-depth studies like PSA, as well as to the results of research, in particular on severe accidents.

Three important general safety objectives have been set up:

- A further reduction of the core melt frequency.

- The "practical elimination" of accident situations which could lead to large early releases of radioactive material. If those situations cannot be considered as physically impossible, provisions have to be taken to "design them out".

- For low pressure core melt situations the design has to be such that the associated maximum conceivable releases would necessitate only very limited protective measures in area and time (no permanent relocation, no need for emergency evacuation outside the immediate vicinity of the plant, limited sheltering, no long-term restrictions in the consumption of food).

A fourth objective can be added: A further reduction of occupational exposure to plant personnel.

The first and second objectives are in line with the present safety concept. The third objective characterises the development of a safety philosophy in the sense of an extension of "defence-in-depth" principles by adding an additional level of defence at the design stage. The technical principles derived from this safety objective ask for new
technical solutions going beyond those which are presently implemented in operating plants for risk reduction (accident management measures).

Since 1994, the common safety approach has been further developed, extended and refined. The subjects which have been treated in this development and refinement process have been selected according to several criteria with an important time priority criterion being the relative date at which an important decision has to be made within the design process. There are also some subjects which are new and which have not yet been addressed in previous licensing practices (e.g. core melt accidents).

This continuous further development of the safety approach can be grouped into two periods. The first period, in which key safety issues with a high time priority have been treated, lasted from 1993 to the beginning of 1995. Main subjects are laid down in GPR/RSK recommendations. They concentrated on

- Severe Accidents (general approach, radiological consequences, impact on containment design, R&D needs,)
- System Design and Use of PSA (general approach, use of PSA, primary system overpressure protection and depressurisation, reliability of shutdown function, consideration of shutdown states, secondary side heat removal, electrical power supplies, residual heat removal, secondary side overpressure protection, secondary side ruptures and containment bypass),
- Internal and External Hazards,
- Primary Circuit Integrity (break preclusion concept),
- Radiological Consequences of Accidents,
- Radiation Protection during Normal Operation.

An overview up to this phase has been given in two conference papers /FRI 95, QUE 95/. In the second phase some of the subjects of the first phase have been and are being further refined (e.g. system design), and new issues have been treated (e.g. internal hazards) and are now treated (e.g. core design). Results are presented in /QUE 97/. Recommendations, requirements and comments developed for these subjects have been continuously issued as GPR/RSK recommendations adopted by DFD. A general overview has also been given at the ARS '97 conference /FRI 97/. The de-
gree of detail within the refinement of the approach is the result of a balance between being sufficiently concrete to give the designer a good guidance and sufficiently general to allow for several design solutions.

Four examples have been selected here to demonstrate both the achievements already obtained and the aim for further safety improvement: System design: Redundancy and Diversity, Severe accidents: Low Pressure Core Melt Situations, High Pressure Core Melt Scenarios.

2.1 Redundancy and Diversity.

This area is an important aspect of safety system design. Not surprisingly, many differences exist in the safety practice of both countries. It is not the general safety philosophy (e.g. implementation of the defence-in-depth principle), but their technical realisation which shows differences, e.g. with respect to degree of redundancy, degree of automation, superposition of events and failures, etc...

Among the principles related to system design are recommendations related to redundancy and diversity:

- The reinforcement of the "defence-in-depth" of the plants will, generally speaking, imply a more extensive consideration of the possibilities of multiple failures and the implementation of diversified means to fulfil the three basic safety functions - reactivity control, cooling the fuel and confining radioactive substances - whatever the state of the plant.

- A reduction of the global probability of core melt will be achieved through an adequate combination of redundancy and diversity in safety systems; on this point, it is underlined that the unavailability of a redundant safety system consisting of identical trains probably cannot be demonstrated to be less than 10⁻⁴ per demand, and that due consideration has to be paid to the support systems when assessing the benefits from diversified systems or equipment.

- For determining the adequate combination of redundancy and diversity in safety systems, the designer can, use probabilistic targets as orientation values; in that case, orientation values of 10⁻⁴ per year for the probabilities of core melt due to
internal events respectively for power states and for shutdown states could be used, having in mind the necessity to consider associated uncertainties.

- For frequent initiating events, the reliability requirement on a safety function is such that two diverse systems or pieces of equipment might be necessary.

- For the safety systems required during all reactor states, including shutdown states, the designer has to combine the single failure criterion with the scheduled maintenance, taking into account the required capacity of the corresponding safety function during this maintenance.

These principles are mainly aiming at a reduction of the core melt frequency. There are additional principles to be followed for systems which contribute to the "practical elimination" of such accident situations which would lead to large early releases.

2.2 Low Pressure Core Melt Situations

According to the third general safety objective, low pressure core melt situations have to be coped with and the corresponding radioactive releases have to be limited. This implies the investigation of various phenomena and the development of a strategy from which the relevant design criteria can be derived. The main goal of this strategy is to identify those phenomena and conditions which have to be considered in the design of the containment and those which have to be eliminated by design.

Concerning the aspects of energy release from hydrogen reactions, the position adopted by GPR and RSK is that the design must cope with a hydrogen production corresponding to the complete reaction of the fuel clad zirconium with water. However, it can be assumed that this amount of hydrogen is not instantaneously generated and released into the reactor building, but as a function of time to be estimated for representative core melt accident sequences.

Catalytic recombiners can be used to limit the concentration of hydrogen in the reactor building, provided the efficiency of such equipment is clearly demonstrated under core melt accident conditions.

The volume of the reactor building and the mitigation means provided must be such as to prevent any possibility of a global hydrogen detonation. Local high concentrations of
hydrogen must also be prevented as far as possible by the design of the internal structures of the reactor building; if it is not possible to demonstrate that the hydrogen local concentration remains below 10 %, specific measures have to be implemented, such as „inertisation“, or reinforced walls.

GPR and RSK have stated that the installation of an internal liner should be considered because it would provide additional margins with respect to containment leak tightness, especially considering specific phenomena such as a fast local hydrogen deflagration. They doubt that for the present designer proposal the defined level of leak tightness can be ensured in view of the expected long-term behaviour of the pre-stressed concrete structure.

Experiments related to the behaviour of composite liners are underway and will be evaluated with respect to the containment leak tightness function.

Concerning the ex-vessel molten core cooling, the strategy adopted by the designer is to spread the molten core over a large area before flooding it with water. In their recommendations of 1994, GPR and RSK have stated that the designer has to propose adequate design provisions to limit the amount of water which could be present in the reactor pit and the spreading compartment at the time of the reactor pressure vessel meltthrough. The possibility of large steam explosion during corium flooding must be prevented and loads resulting from melt/water interaction must be taken into account in the design.

Further studies show that these objectives could be achieved by a large spreading „dead-end“ compartment spatially separated from the reactor pit and protected from the thermo-mechanical loads consecutive to the reactor pressure vessel failure. Design provisions would prevent the flow of condensate from any part of the containment into this compartment. Moreover, a steel gate would physically separate the reactor pit from the spreading compartment. In this concept, scarificial concrete layers would be implemented in the reactor pit and in the spreading compartment to obtain adequate characteristics of the melt. The basemat penetration would be prevented by a protective refractory layer covered by a cast iron layer. The cooling of the melt would be ensured by melt flooding from above by water coming from the large water tank inside the containment building associated to the containment heat removal function. Thermal
loads on the basemat would be limited by a cooling device below the protective refractory layer linked to the containment heat removal system.

However, GPR and RSK underline that the validation of such a strategy would require extensive research and development work. The robustness of the concept described above has to be checked for various scenarios, including late reflooding and low residual power scenarios; specific attention has to be paid to the possibility of an early or partial failure of the steel gate as well as to the optimisation of the reactor pit design, in terms of composition and masses of the sacrificial concrete layers and of the transfer channel between the reactor pit and the spreading compartment. The refractory layer behaviour has also to be validated taking into account the capacities of the cooling systems and the possibilities of thermo-mechanical attacks by iron oxides or corium oxides.

Specific provisions have also to be implemented to ensure that the reactor building basement would remain leak-tight in order to prevent contamination of soil and groundwater.

With respect to containment heat removal, GPR/RSK emphasise the fact that this function must be ensured without a containment venting device. In addition, adequate provisions have to be foreseen to obtain a short-term reduction of the containment pressure.

The objective of the limitation of radioactive releases implies a substantial improvement of the containment function, considering the different possible failures of this function during core melt situations. This improvement can be achieved through the use of a double-wall containment concept, and an annulus between the inner and outer walls being maintained at a subatmospheric pressure in order to collect the releases of all possible leaks through the inner wall and to filter them before they are released to the environment via the stack.

The design pressure and temperature of the containment inner wall must be such as to allow a grace period of at least 12 hours without containment heat removal and to ensure its integrity and leak tightness even after the global deflagration of the maximum amount of hydrogen which could be contained in the containment building in the course of low pressure core melt accidents.
The limitation of radiological consequences of low pressure core melt accidents is specified in general terms in the third safety objective of chapter 3.1. The numerous aspects of the required analyses and assumptions have been treated in detail, supported by analyses performed by GRS and IPSN. More details are presented in /BAC 97/.

2.3 High Pressure Core Melt Scenarios

The second safety objective asks for the „practical elimination“ of accident situations which could lead to large early releases of radioactive material. One of these situations is a vessel melt-through at high primary system pressure.

In the „Common Safety Approach of 1993“ it is stated that: „It must be a design objective to transfer high pressure core melt sequences to low pressure core melt sequences with a high reliability so that high pressure core melt situations can be „excluded“.“ The above mentioned design objective can be dealt with by means of pressure relief valves at the primary circuit, with such a discharge capacity to limit the pressure in the reactor coolant system in the range of 15 to 20 bar at the moment of the reactor pressure vessel failure. The designer has to propose such means with due consideration of the expected reliability of the valves; in particular these means must be clearly qualified under representative conditions. As an orientation, the equipment used to depressurise the primary circuit has to be as reliable as the relief valves used to prevent an overpressure. The use of specific valves - to be actuated only in case of core melt sequences - should be investigated.

The EPR project has proposed a design solution with a dedicated bleed valve for primary system depressurisation in case of a failure of the pressuriser valves, which are supposed to be used for depressurisation as a first choice. On this solution more detailed GPR/RSK comments have been given recently:

Their discharge function must be available in case of loss of off-site power and unavailability of all diesel generators. Once open, the bleed path should stay fully open with high reliability through the progression of the accident.

The discharge capacity of the dedicated valve has to be determined considering the following scenarios, with realistic assumptions:
• Loss of off-site power with unavailability of all diesel generators,

• Loss of off-site power with unavailability of all diesel generators, but with recovery of water supply during core melting,

• Total loss of feedwater and failure of primary feed and bleed.

Sensitivity studies regarding the discharge capacity, the hot gas temperature and the initiating criteria have to be done considering delayed bleeding and late reflooding as well as the uncertainties of the code models related to the late core degradation phase or reflooding.

In addition, design provisions have to be taken to cope with the mechanical loads which would result from the reactor vessel failure at 20 bar so as to limit the vertical upward movement of the pressure vessel. Design provisions are also necessary to ensure that large quantities of corium released from the reactor pressure vessel cannot be carried out of the reactor pit.

3 Future Steps in the Development of the Safety Approach

The French-German harmonisation concerning the safety requirements for future pressurised water reactors has made considerable progress as has been demonstrated in this paper by means of some selected issues. Further achievements have been obtained in many areas not explained here. The process is further advancing.

Conclusions on some essential issues are still pending. Further information is necessary before concluding on the general design of the safety systems. It should be borne in mind that this design must, indeed, be examined with respect to the two essential objectives: reduction of the core melt frequency and "practical elimination" of accident situations liable to lead to large early releases of radioactive substances. Furthermore, in view of the results of probabilistic safety assessments for existing plants, this examination should cover shutdown situations for which there is no doctrine equivalent to that developed in the past for situations with the reactor at power.

An important area is the development of accident analysis rules. Common guidelines are under consideration which include the technical acceptance criteria (decoupling
criteria) and the methodology of accident analysis. In 1996 GPR/RSK already stated that realistic assumptions and models could be used for the safety demonstration of LOCA. The compliance of the results with the acceptance criteria must be proven at a high confidence level which implies the use of a qualified and validated frozen code version and an explicit evaluation of the associated uncertainties by combination of the elementary uncertainties (code models, scaling effects, initial and boundary conditions, user effects). GPR/RSK would also accept as an alternative approach the use of models and criteria already applied to existing plants in a conservative way.

Another area not yet completed is severe accidents and containment design. Some conclusions can be drawn only after results of experimental programmes are available. Related to this subject is the question whether the necessary containment leak tightness (1 Vol.% per day) can be demonstrated. GPR/RSK has recommended to investigate an inner liner attached to the inner wall of the double wall containment (see chapter 2.2).

Some subjects have not yet been treated within the harmonisation process, because they are not yet needed in the design process, such as instrumentation and control.

So far, the common activities have resulted in a substantial number of common GPR/RSK recommendations, adopted by DFD. The aim was to arrive at common agreements without national deviations. The set of all agreements is supposed to be a strong guidance for the continuation of the designers' work and within potential future licensing processes in France and Germany. The legal implementation in the licensing process may be different in the different countries, however, their technical content is not meant to be subject to individual changes.

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