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NUCLEAR ENERGY AGENCY
COMMITTEE ON THE SAFETY OF NUCLEAR INSTALLATIONS

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Advanced Thermal-hydraulic and Neutronic Codes: Current and Future Applications

Summary and Conclusions

OECD/CSNI Workshop
Barcelona, Spain
10-13 April, 2000

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ORGANISATION FOR ECONOMIC CO-OPERATION AND DEVELOPMENT

Pursuant to Article 1 of the Convention signed in Paris on 14th December 1960, and which came into force on 30th September 1961, the Organisation for Economic Co-operation and Development (OECD) shall promote policies designed:

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

The original Member countries of the OECD are Austria, Belgium, Canada, Denmark, France, Germany, Greece, Iceland, Ireland, Italy, Luxembourg, the Netherlands, Norway, Portugal, Spain, Sweden, Switzerland, Turkey, the United Kingdom and the United States. The following countries became Members subsequently through accession at the dates indicated hereafter: Japan (28th April 1964), Finland (28th January 1969), Australia (7th June 1971), New Zealand (29th May 1973), Mexico (18th May 1994), the Czech Republic (21st December 1995), Hungary (7th May 1996), Poland (22nd November 1996), Korea (12th December 1996) and the Slovak Republic (14 December 2000). The Commission of the European Communities takes part in the work of the OECD (Article 13 of the OECD Convention).

NUCLEAR ENERGY AGENCY

The OECD Nuclear Energy Agency (NEA) was established on 1st February 1958 under the name of the OEEC European Nuclear Energy Agency. It received its present designation on 20th April 1972, when Japan became its first non-European full Member. NEA membership today consists of 27 OECD Member countries: Australia, Austria, Belgium, Canada, Czech Republic, Denmark, Finland, France, Germany, Greece, Hungary, Iceland, Ireland, Italy, Japan, Luxembourg, Mexico, the Netherlands, Norway, Portugal, Republic of Korea, Spain, Sweden, Switzerland, Turkey, the United Kingdom and the United States. The Commission of the European Communities also takes part in the work of the Agency.

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.

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

COMMITTEE ON THE SAFETY OF NUCLEAR INSTALLATIONS

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

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

10-13 April, 2000
Barcelona, Spain

SUMMARY AND CONCLUSIONS

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OECD/CSNI WORKSHOP ON
ADVANCED THERMAL-HYDRAULIC AND NEUTRONIC CODES:
CURRENT AND FUTURE APPLICATIONS

EXECUTIVE SUMMARY

An OECD Workshop on Advanced Thermal-Hydraulic and Neutronic Codes Applications was held from 10th to 13th of April 2000, in Barcelona, Spain, sponsored by the Committee on the Safety of Nuclear Installations (CSNI) of the OECD Nuclear Energy Agency (NEA). It was organised in collaboration with the Spanish Nuclear Safety Council (CSN) and hosted by CSN and the Polytechnic University of Catalonia (UPC) in collaboration with the Spanish Electricity Association (UNESA).

The objectives of the Workshop were to review the developments since the previous CSNI Workshop held in Annapolis [NEA/CSNI/R(97)4; NUREG/CP-0159], to analyse the present status of maturity and remnant needs of thermal-hydraulic (TH) and neutronic system codes and methods, and finally to evaluate the role of these tools in the evolving regulatory environment.

Following introductory remarks by L. Echávarri, Director General of OECD/NEA, two keynote presentations were given by Prof. A. Alonso (CSN Counsellor) and Dr. Nils J. Díaz (USNRC Commissioner). The Workshop closure included the participation of M. Livolant, Director of IPSN and Chairman of CSNI.

The Technical Sessions and Discussion Sessions covered the following topics:

- Regulatory requirements for Best-Estimate (BE) code assessment
- Application of TH and neutronic codes for current safety issues
- Uncertainty analysis
- Needs for integral plant transient and accident analysis
- Simulators and fast running codes
- Advances in next generation TH and neutronic codes
- Future trends in physical modeling
- Long term plans for development of advanced codes

The focus of the Workshop was on system codes. An incursion was made, however, in the new field of applying Computational Fluid Dynamic (CFD) codes to nuclear safety analysis.

Findings and Recommendations

Keynote speakers addressed the issue of code development and application to achieve and maintain the highest safety, as well as sound competitiveness of the nuclear installations.

Maturity of TH codes is recognised, for example in the USNRC Code Consolidation Program, and further development is considered necessary only in specific areas. Warnings were expressed that code development could be hampered by lack of experimental results, difficulties in data preservation and human capital depletion.

Best-Estimate (BE) analysis, supplemented by evaluation of uncertainty, is considered to play a dominant role in risk informed regulation whether or not the licensing basis is affected. In either case the determination of a success criterion (PSA terminology) or a margin to a safety limit should rely on a realistic approach.

Coupling of TH/neutronic codes should be extended to fuel behaviour codes (mainly for fast transients). This raises the question of uncertainty propagation between coupled codes. In particular, the assessment of uncertainty in 3D kinetics should be emphasised; this need is made more urgent by the high burn-up issue. Real plant transients and benchmarking seem to be needed to attain this goal.

Connectivity among codes calls for a generalised coupling protocol. Otherwise code communication becomes a bottleneck for any project. A communication standard would help to facilitate coupled code development.

Quality Assurance (QA) tools and ancillary aids like graphical user interfaces need to be given sufficient importance for BE applications. QA allows for a better control of the user effect and also helps to gain confidence in the methodology (code, model, application) and the supporting organization.

Advances in numerics specially related to 2-D and 3-D developments need to be benchmarked. Progress since Annapolis has been achieved, but there still exists a need to validate different approaches by means of adequate benchmarking. Steps in this direction should be endorsed.

Current developments in the field of interfacial area density transport should be extended to all flow regimes and transitions between them. Experiments in support of these developments should be established.

Computational Fluid Dynamics (CFD) application will rely on further development of user aids and extension to two-phase regimes. Current CFD code applications cover situations where the detailed study of local behaviour is required such as thermal stratification or boron dilution.

Advanced numerical simulation techniques will help in developing closure relations; they may provide information complementing experimental data.

BE code application comprises many different areas. The maturity achieved allows for cautious introduction of BE based methodologies in the licensing process. A stimulating example of how to determine system code accuracy was presented; this way code validation subjectivity can be reduced. Further international co-ordinated efforts among different organisations are recommended.

The evaluation of event trees in PSA requires realistic assumptions and BE analytical tools. The viability of PSA and BE code coupling to dynamically simulate accident scenarios has been shown. Verification of the added value of this technique against current static applications would give rise to a promising field of application.

Conclusion

As a general conclusion, the Barcelona Workshop can be considered representative of the progress towards the targets marked at Annapolis almost four years ago. The Annapolis Workshop had identified areas where further development and specific improvements were needed, among them: multi-field models, transport of interfacial area, 2D and 3D thermal-hydraulics, 3-D neutronics consistent with level of details of thermal-hydraulics. Recommendations issued at Annapolis included: developing small pilot/test codes for new numerical methods, benchmarking physical models, preserving and enlarging the experimental database for code validation, and improving codes' "user friendliness".

The Barcelona Workshop has reviewed the development since then. It observed that the evolution is in line with those recommendations and that substantial progress was made. Additional recommendations were issued. The basic trends from Annapolis still hold and are reaffirmed in the frame of evolving regulatory and commercial environments.

OECD/CSNI WORKSHOP ON
ADVANCED THERMAL-HYDRAULIC AND NEUTRONIC CODES:
CURRENT AND FUTURE APPLICATIONS

MEETING SUMMARY

1. Sponsorship

An OECD Workshop on Advanced Thermal-Hydraulic and Neutronic Codes Applications was held from 10th to 13th of April, 2000, in Barcelona, Spain, sponsored by the Committee on the Safety of Nuclear Installations (CSNI) of the OECD Nuclear Energy Agency (NEA). It was organised in collaboration with the Spanish Nuclear Safety Council (CSN) and hosted by the CSN and the Polytechnic University of Catalonia (UPC) in collaboration with the Spanish Electricity Association (UNESA).

2. Background of the Workshop

On 6-8 November, 1996, the OECD Workshop on Transient Thermal-Hydraulic and Neutronic Codes Requirements was held in Annapolis, USA [NEA/CSNI/R(97)4; NUREG/CP-0159]. The issues raised during the meeting included current and future uses of thermal-hydraulic and neutronic codes, additional experimental needs, numerical methods and requirements for future generation of safety analysis codes. Some of the questions addressed were:

- What capabilities do the users want for the next 10 years?
- What code features should be provided to assist users?
- What should the user interface be like, considering both the front-end and the back-end of the analysis process?

The Annapolis Workshop resulted in a number of conclusions and recommendations. Specific areas identified for improvements include multifield models, transport of interfacial area, 2D and 3D thermal hydraulics and 3-D neutronics consistent with level of details of thermal hydraulic models. Other topics discussed in detail are numerical methods and features, issues of modeling approach and technology, user needs as well as needs for experimental data base for codes validation. In the area of, for example, modeling methodology, recommendations include: developing small pilot/test codes to test new methods, and directing research into developing and benchmarking physical models. It was recognised that there is a need for better user interface modules to improve codes' "user friendliness".

Next, there was the Second CSNI Specialist Meeting on Simulators and Plant Analysers held in Espoo, Finland, on 29 September - 2 October, 1997 [VTT Symposium 194]. One of the issues discussed was the feasibility of developing a software suitable for performing plant safety analysis based on advanced thermal-hydraulic and neutronic codes.

Also, the CSNI International Seminar on Best Estimate Methods in Thermal-Hydraulic Safety Analyses was held in Ankara, Turkey, on 29 June - 1 July, 1998 [NEA/CSNI/R(99)22, NEA/CSNI/R(99)10]. One of the major concerns raised at the Seminar was the level of confidence in Best Estimate (BE) codes and in the evaluation of codes' associated uncertainties. One of the general recommendations is to further promote the applications of BE approach, with reliable evaluation of uncertainties, as a basis for evaluation of the plant safety limits and existing additional safety margins, forming full scope analysis. It was also recognised that studies of beyond DBA and severe accident scenarios are very important to evaluate a level of "enhanced' plant safety.

These and other CSNI activities in the area of evaluating plant safety margins led to the conclusion that there was a need for another exchange of information on the progress made in developing advanced thermal-hydraulic and neutronic codes and their application to plant safety analyses.

3. Scope and Technical Content of the Workshop

The scope of the Workshop included:
- review of progress made since the Annapolis meeting in modeling improvements such as inclusion of interfacial area transport field, full 3D neutronics and thermal hydraulics, etc.
- progress made in codes' verification & validation process and numerical benchmarking.
- adequacy of advanced codes for use in support of current safety related issues, e.g., PTS and RIA,
- benefits of applying the advanced codes and methodologies to improve plant performance while maintaining or even enhancing its safety,
- current and future needs and/or requirements for the advanced codes,
- interest of Regulatory Authorities in advanced BE methodologies.
- coupling with neutronic, containment, SA, PSA and advanced reactors designs codes and models,
- computational fluid-dynamic techniques (CFD codes).

The Technical Content was arranged in several topics spanning the scope of the Workshop included:

Topic 1: Long term plans for development of advanced codes.
- accomplishments since Annapolis 1996 meeting.
- experimental needs.

Topic 2: Regulatory requirements for BE code assessment, including:
- risk informed environment,
- strategies to allow for BE applications.(deterministic vs. probabilistic)
- measures to ensure high quality application.

Topic 3a: Overview of applications of TH and neutronic codes for current safety issues.
- exemplify specific needs derived from safety or regulatory concerns.

Topic 3b: Needs for integral plant transients and accident analyses. Known limitations.
- validation needs.
- data base and human capital preservation
- experience with applications
- new generation reactors.

Topic 4: Advances in next generation of TH and neutronic codes.
- implementation of pre/postprocessors
- numerical techniques
- modularity and coupling interfaces.

Topic 5: Future trends in physical modelling for next generation codes:
- 1D and 3D modeling,
- CFD codes.
- phase flow developments
- special components modeling.

Topic 6a: Uncertainty analysis, level of confidence.
- experience with applications.
- embedded uncertainty

Topic 6b: Simulators and fast running codes.
- Fields of applications
- Speed versus confidence
- Programming and computer performance.

A technical session was dedicated to each topic. Every day a technical discussion session was held to deal with the relevant issues of the day's sessions. At the end of the workshop three parallel sessions were held to address such important areas as: i) Coupled TH/Neutronic codes; ii) Use and applications of BE codes; iii) Future R&D in TH modeling and numerics. The conclusions from these sessions were presented at a plenary session. Each group discussion was facilitated by internationally respected experts.

In general terms, the Workshop attempted to give answers to questions in the following areas:

- What are the benefits of using coupled TH/neutronic analysis?
- What are major technical issues and challenges associated with applications of advanced coupled TH/neutronic codes?
- What are accuracy requirements for the current and future generations of reactor safety codes?

4. <u>Program Committee of the Workshop</u>

A Program Committee (PC) was nominated by the CSNI to evaluate the abstracts of proposed papers, to select the papers for presentation, to organise the Sessions and to develop the final program of the workshop, appoint Session Chairmen, etc. Its members were:

- Prof. Agustín ALONSO, General Chairman of the Meeting, CSN, Spain
- Mr. Nusret AKSAN, PSI, Switzerland,
- Dr. José Mª ARAGONÉS, NSC liaison , Polytechnic University of Madrid, Spain
- Dr. Dominique BESTION, CEA, France
- Prof. Francesco D'AURIA, University of Pisa, Italy
- Dr. Mamdouh EL-SHANAWANY, HSE , UK
- Dr. Farouk ELTAWILA, NRC, USA
- Dr. John LUXAT, Ontario Hydro, Canada
- Dr. Riitta KYRKI-RAJAMÄKI, VTT, Finland
- Mr. Fernando PELAYO, CSN, Spain (<u>PC Chairman</u>)
- Dr. Francesc REVENTOS, ANACNV/UPC, Spain
- Mr. Victor TESCHENDORFF, GRS, Germany
- Mr. Miroslav HREHOR, NEA (PWG2 S<u>ecretary</u>)

The Workshop was attended by some 120 participants from 21 countries from the OECD and others. The list of participants is given at the end of the proceedings.

5.0 Summary structure.

The document begins with a summary of each technical paper made by the session chairmen. Every two technical sessions (corresponding to a workshop journey) a summary of the day's open technical discussion is presented. This is designed so that the conclusions on: Consolidated Achievements, Limitations and Development Needs are presented and in some cases ranked high (H), medium (M), low (L). This approach is repeated for the final parallel sessions.

KEYNOTE PAPERS SUMMARY

Chairman: F. Pelayo

Keynote paper 1: **"The use of Thermal-Hydraulic and Neutronic Advanced Codes in Intermediate Countries: The case of Spain",** Prof. A. Alonso, Counselor of CSN, Spain

Prof. Alonso's presentation addressed the role played by the so-called exporter versus qualified importer countries on the development and refinement of advanced codes, all of it resulting in common benefit. Through a historical analysis spanning early developments and a maturity phase, it is pointed out how the collaborative effort exemplified by several international projects, acted in a synergetic manner to consolidate international cooperation, helping intermediate countries to gain recognition and keeping pace with developments in the field. The paper reaffirms the need for maintaining collaboration in the future at national, incorporating all interested parties, and international level as a means to bring together experts from different countries facilitating the development of better, more economical computation methods. Finally the new economy environment affecting all countries is pointed out as a driving force to work on a basis of a more efficient regulation which in turn requires an enhanced international collaboration to help regulators to reduce unnecessary burdens.

Keynote paper 2: **"Heat and Heat Sinks; Safety and Cost Competitiveness",** Dr. Nils J. Diaz, Commissioner, US NRC

Dr. J. Nils Díaz, under the title of his presentation, points to the correlation between safety and competitiveness. Dr. Diaz believes that one needs to focus the regulator's and the nuclear industry's attention and resources on what is relevant for safety. With this approach, safety and cost competitiveness can become compatible and enabling factors. By means of proper management of plant safety, enhanced plant economic performance can be achieved. Turning to thermal-hydraulics and neutronics, according to the author, the reduction of unnecessary conservatism by means of best estimate analytical tools in the context of risk-informed regulation should enhance the value of nuclear technology to society by improving safety and cost competitiveness. The challenge is cast to the audience by the author in terms of "how best to systematically use state of the art thermal-hydraulics and neutronic codes to improve safety and cost competitiveness and lays the foundation for a new generation of nuclear power plants".

TECHNICAL SESSION 1: Long Term Plans for Development of Advanced Codes.

Session Chairmen: M. Réocreux, N. Aksan

The main aim of the Technical Session 1 was to present a general overview on the developments and applications of advanced thermal-hydraulics and neutronic codes since the Annapolis Meeting in 1996 and also to provide the objectives of the present meeting. This objective was covered with the first paper of the session. In addition, there were presentations from four major countries on their present work and long term plans on their advanced thermal-hydraulic code development programmes.
Summaries of the five papers presented in this session are given below:

1. Evolution of Developments and Applications of Advanced Thermal-hydraulic and Neutronic codes, M. Réocreux (IPSN) and N. Aksan (PSI)

As the first paper of the technical sessions in the Workshop, this paper provides the general background information and very brief summaries of the earlier two Workshops, which introduce the starting point information to this Workshop.

After a historical review of OECD/CSNI Specialist Meetings on Two Phase Flow, the status of thermalhydraulics and coupled neutronic codes in 1996 and recommendations on the directions for future actions, as concluded at the Annapolis Meeting, were presented. These actions and conclusions covered the physical modeling, the numerical methods, modeling methodology issues and user needs. The status on the use of the Best Estimate codes as covered in 1998 at Ankara Seminar and conclusions of this Seminar are also given in summary.

As a result of these summaries, the objectives and frame for the “OECD/CSNI Workshop on Advanced Thermal-Hydraulic and Neutronic Codes: current and Future Applications” are put in perspective. A number of questions were asked in order to review the progress made during the four years since the Annapolis Workshop. The answers to these questions during this Workshop will be contributing to the planning of the future developments of the advanced thermalhydraulic and neutronic codes, as well to the solutions of several challenging key items.

2. US NRC code consolidation and development effort, J. Uhle (NRC) and B. Aktas (Scientech)

The US NRC is currently consolidating the capabilities of its thermal-hydraulic codes into a single code in order to reduce the maintenance and development burden. The details of the consolidation plan, the achieved results to date were presented covering the areas of the code architectural improvements (code language, data base design, code modularity), general modeling capabilities (e.g., stability, 3-D kinetics capabilities of TRAC-B and RELAP5 codes, etc.), code improvements (e.g., graphical user interface, exterior communication interface, subcooled boiling at low pressure, interfacial area transport, phase separation at tees, rod bundle heat transfer).
Currently, work focuses on assessing the consolidated code TRAC-M in order to identify the TRAC-B constitutive models to be integrated into a consolidated code. Once the BWR assessment is complete, the same approach will be utilized to consolidate the RELAP5 capabilities. With the use of a consolidated code, the user community is expected to focus on one code instead of four and code improvements will be made more efficiently.

3. The French program of CEA, IPSN, EDF, and FRAMATOME for the next generation of thermal hydraulic codes, D. Grand (CEA), M. Durin (IPSN), L. Catalani (FRAMATOME), J-P. Chabard (EDF)

This paper mainly deals with the proposal for thermal-hydraulic research applied to nuclear reactors in France. After an evaluation of the challenges in terms of the competitiveness and safety of nuclear energy, areas are identified where increased knowledge and advances in thermal-hydraulics are expected. They are in two complementary directions: The improvement of the two-fluid model by the inclusion of additional transport equations and the development of the simulation of the fine scales of the flow. The greater computing efficiency is expected to be fully used, if the modeling is enhanced. Recommendations are also made for the development of instrumentation and definition of new experiments. An outline is given for the progression of the program from present day tools to future tools based on advanced two-fluid model and modeling of local phenomena.

4. The Industry Standard Toolset (IST) of codes for CANDU safety analysis, J. C. Luxat (OPGI), W. Kupferschmidt (AECL), P. D. Thompson (NBP), M-A. Petrilli (H-Q)

A number of projects to upgrade the quality of safety analysis software have been undertaken by the Canadian nuclear industry. During the establishment of these projects, it was recognized that developments to enhance the capabilities, verification, validation, qualification nd, maintenance of these codes represent a large commitment of resources from organizations with the industry. In order to reduce redundant work in different organizations a consolidated common set of computer codes for safety analysis, referred to as the Industry Standard Toolset (IST) initiative, has been initiated and the principles and organization of this initiative are presented, including principles, scope, management structure and process. Work is proceeding in this area in Canada to complete code qualification by the end of 2001.

5. Status and plans for future development of thermal-hydraulic and neutronic codes in Germany, V. Teschendorff (GRS), F. Depisch (SIEMENS)

Germany actively develops and applies thermal-hydraulic and neutronic codes. The development of the thermal-hydraulic system code ATHLET includes dynamic flow regime modeling, a 2D/3D module, and improved time integration for code speed-up. The validation process of this code will be completed and the remaining uncertainties quantified based on GRS's own methodology. For 3D neutron kinetics, the codes QUABOX/CUBBOX and DYN3D are available, both coupled to ATHLET for analysing transients with strong interaction between core and coolant system. The performance of the coupled code system was demonstrated by international benchmarks and various applications. SIEMENS has separately developed the code system CASCADE-3D for core analysis. For coupled transients, it can be linked with S-RELAP5 which is an in-house developed 2D code based on RELAP5.
CFD codes are already successfully applied for detailed 3D computation of single phase mixing problems. It is planned to launch a development of two-phase capabilities for CFD methods.

TECHNICAL SESSION 2: Regulatory Requirements for BE Codes Assessment.

Session Chairmen: **M. El-Shanawany, J. Uhle**

The aim of Technical Session 2 was to present the recent advances in the “Best Estimate” safety analysis. The Session also proposed and identified some areas of work, which need to be developed in order to gain acceptance by both the power plant operators and the regulatory bodies.
Four papers were presented covering a wide range of topics such as “risk informed regulatory approach" and a proposed set of practical suggestions for performing safety analysis.

1. Change to a Risk Informed Regulatory Framework at USNRC, Challenges and Achievement

After significant interactions with the US industry, the USNRC has formulated a strategy for change utilising a Risk-Informed regulatory approach. The guidelines permit plant changes which results in a very small increase in core damage frequency and large early release frequency for the plant, provided that the licensees track and the NRC staff monitors the accumulative effect of such changes. In order to conclude that the safety mission is fulfilled, the NRC has also developed a new risk-informed structured process to oversee the performance of nuclear power plants. This process monitors licensee performance in three broad areas, namely ; reactor safety, radiation safety and plant safeguards.

Discussion : With reference to thermal hydraulics, the NRC indicated that all the aspects of Appendix K, including thermal hydraulics acceptance criterion, are being reviewed with the aim to reduce the regulatory burdensome, but without compromising the safe operation of the plants.

2. A Regulatory Approach to Assess Uncertainties Treatment for Licensing Purpose

The Belgium Regulatory Body (AVN) presented a novel approach to assess the uncertainty treatment in licensing calculations. It is based on a set of rules aimed at identifying acceptable combinations of uncertainties covering both systematic and random errors. In order to reach an informed decision, emphasis is placed on requesting information about both the sources of uncertainties and the models used in performing the licensing calculations.
A procedure has been presented to practically apply these principles. It is based on subdividing the model into a number of basic elements.

Discussion : It was clear that more cases needed to be analysed using this proposed novel approach to demonstrate its merits. Furthermore, a comparison with current uncertainties methods would be an advantage in terms of understanding the details and the importance of the different aspects of this approach.

3. Current and Future Application of Thermal-Hydraulic and Neutronic Best- Estimate Methods in Support of the Swiss Licensing Process

This paper proposed an alternative method to that of the USNRC´s CSAU uncertainty approach, which required a significant amount of analysis to be carried out. The proposal is based on performing best estimate analysis with bounding input parameters combined with sensitivity calculations to quantify the relevant bounding input parameters. Two examples of the proposed approach were presented, PWR with reduced availability of ECCS and MSIV-ATWS event in a BWR. The paper concluded that, although the proposed best estimate approach seems to be useful, a comprehensive systematic assessment of the analysis uncertainty is still lacking. Future work will include 3D kinetics, assessment of high-burnup fuel performance in RIA and LOCA conditions.

Discussion : The main point to note is that best estimate analysis should be looked at as a package of work utilising a number of aspects such as ;

- best estimate codes. The codes shortcomings must be will characterised and well
- understood and taken into account in performing plant safety analysis.
- the critical parameters affecting the calculations need to be identified.
- the uncertainties in the plant's operating boundary conditions need to be understood.
-

4. IAEA Safety Report on Accident Analysis for Nuclear Power Plants

This paper presented a summary of the IAEA´s report on Accident Analysis of Nuclear Power Plants. The report is aimed mainly at the developing countries. The objective of the report is to establish a set of practical suggestions based on the best practice worldwide for performing deterministic safety analysis. It covers the selection of initiating events, acceptance criteria, computer codes, modeling assumptions, preparation of input, qualification of users, presentation of results and quality assurance.

OPEN TECHNICAL DISCUSSION SESSIONS 1 & 2

Chair: M. Réocreux
Panelists: N. Aksan, J.L.Uhle, M. El-Shanawany

The first **Technical Session** was mainly informative and provided a review on the long-term plans for the development of advanced codes. The presentations often referred to more detailed papers, which were presented later. Because it was foreseen to have detailed discussions in the following sessions, the questions discussed in the present open session have been restricted to requests on code assessment in the overall planning.

Consolidated achievements.

For the US NRC, we had an extensive and precise view of the work, which is in progress, and also of the work planned up to 2002-2003. For France, the presentation was focused on the advanced thermalhydraulic research activities including the motivations, which support such a program and the plans until 2010 in incorporating results in the next generation two-phase thermalhydraulic codes. This is in parallel with the ongoing work on the present codes CATHARE and SCAR. Then we had a presentation on the Canadian efforts to standardise and harmonise the different codes used in safety analysis. After a short recall of the situation in Germany in relation to nuclear reactors, the German plans to have up to date tools in the near future were presented.

Development needs.

It was specified, for the US NRC plans, that the consolidated code planned for 2002-2003 will capture all the capabilities of the current suite of codes including their assessment. For the following years, there may be deficiencies, which will be identified, but as they will concern one single code, they should be improved at a more rapid rate. It was also expressed that the user feedback may be included in the planning, as past experience showed that it took several years after the release of a code to have it stabilised, taking into account all the questions raised by the users. Consequently, the dates of 2002-2003 may be delayed by 3 to 4 years in order to get a complete consolidation.

Current limitations.

It was also emphasised more generally that code assessment relies essentially on experimental results. This means that the data should be preserved and be kept available through data banks, where the validation matrices should be stored, and this means also that for new physical models the experimental programs which will be needed should be started sufficiently in advance to provide their results in time.

The second **Technical Session** dealt with the points related to the best-estimate codes and their assessment within the safety analysis and regulatory requirements.

Consolidated achievements.

The US NRC presented its risk informed approach in licensing, the objectives of which are to provide alternatives to licensing requirements and to eliminate unnecessary and ineffective regulations. In the second paper, Belgium gives the results of the assessment of the treatment of uncertainties for licensing purposes. This was based on comparative studies, which were performed between the evaluation of instrumentation uncertainties in experiments, and the determination of code uncertainties. Handling this uncertainty question properly is mandatory for the licensing process, when using best-estimate codes.

Switzerland, as an example of a small country case, presented its use of best-estimate codes in the licensing process. This present use was directed to selected applications after the code has been validated on experiments or plant transients. In addition, it has been emphasised that it should also comprise uncertainty evaluation in the future. The last paper in this session was given by the IAEA. It summarises the contents of a report entitled "Accident Analysis of Nuclear Power Plants" and deals with all that should be done in performing deterministic safety analysis.

Current limitations.

The topics of this second session were extensively discussed during the open session. Points, which were discussed, were comprised of:

- Use of Appendix K,
- Best-Estimate codes with uncertainty evaluation,
- Risk informed regulation.

Several statements were made during the discussion, which were often influenced by the particular position of utilities and the regulatory authorities in each individual country. The most representative of these statements can be summarised as follows:

- Concerning the use of appendix K, it appears that it is still largely used by the utilities. The position of the regulatory authorities towards the use of the Best Estimate approach to replace appendix K is in many countries a waiting attitude. The initiative should come from the utilities, which often prefer the comfortable framework of appendix K.
- In the current plant already designed with the old rules, there is no request to change the framework of appendix K, except when some new problems appear in plant operation. Examples were given such as hot leg streaming, new fuel management. In those cases, the need to use less margins pushes towards the use of best or better estimate approach.
- The contribution of the risk informed approach could be, first, to modify or revise appendix K by using some risk insights. As a first step best estimate analysis, as opposed to Appendix K analysis, can be performed without changing the current licensing basis of accidents and transients, which should be considered. A second step could be an entire redefinition, based on risk analysis, of the list of events the plant should be able to cope with. As a consequence of this redefinition the large break LOCA would then probably loose its status of design basis accident.
- The issue of the large break LOCA used as design basis accident (DBA) and the relation between this DBA and appendix K was extensively discussed during the session. It was expressed that substituting large break LOCA by some other accidents for DBA requires normally that the substitute should be defined. Already for new plants the severe accidents have often to be considered and this induced more severe limitations on some parameters than the classical large break DBA does. Nevertheless, it was also considered that any change should be checked carefully as far as large break LOCA serves now as an input to a multitude of design consideration. Going on to the subject of severe accidents, the problem was also raised of the pertinence of the severe accident codes, which was questioned by many participants.

Development needs.

- If the risk informed approach seems to ask for the use of best estimate calculations because it looks contradictory to the appendix K conservative approach, it was also observed that there are different levels in the possible changes of the safety approach. The first one is to benefit of having a better knowledge of the physics by using best estimate calculations with uncertainty evaluation in the accident prediction. This "physical prediction" replaces in fact the conservative margins of appendix K, which were defined to take into account the unknown phenomena at the time Appendix K was promulgated.. This possibility has been already introduced in the modifications of appendix K, which have been defined at the end of the 80's. The next level concerns the assumptions which are made on the plant state and which are used as boundary conditions for the calculations. Many of these assumptions come from safety rules such as the defense in depth or the single failure criteria, which were edicted for the safety approach. Looking at the probability, for instance, of a large break LOCA which is quite low and to which you apply the single failure criteria, this leads to an accident of very low probability. Consequently, the risk informed approach could consider this accident as unimportant, whereas the study of this accident results in fact presently from the basic principles of the deterministic safety approach. Here is certainly the key question, which should be answered for the ultimate use of the risk informed approach.

- In their final conclusions, the panelists underlined that appendix K is still widely used. This situation does not exhibit a significant change since the Ankara meeting. One should be aware that the best estimate approach is in fact an entire package which includes better codes, validation, uncertainties evaluation, estimation of shortcomings and a lot of sensitivity analysis. Risk informed regulation is certainly a powerful tool which relies on PSA / PRA. This means that a continuous revision process should take place for taking into account changes in the plants. It appeared to some participants that risk informed approach should be introduced in a systematic slow process so as to make sure that all the consequences are well understood. This introduction requires also assessing carefully the database, which will be used for the risk data.

TECHNICAL SESSION 3A: Overview of Applications of Thermal-Hydraulic and Neutronic Codes for Current Safety Issues.

Session Chairmen: Prof. J.M.Aragonés, R. Kyrki-Rajamäki

In this session seven papers were presented on various applications of thermal hydraulic codes in most cases coupled with neutronics. The three-dimensionality either or both in neutronics and thermal hydraulics modeling was the common thread of the papers. The new possibilities and limitations given by 3D modeling were studied with comparisons, sensitivity studies, and real applications in important safety issues.

The paper by **Ivanov et al.,** ***OECD/NRC MSLB Benchmark – A Systematic Approach to Validate Best-Estimate Coupled Codes Using a Multi-Level Methodology***, describes the international effort to validate the coupled neutronic thermal hydraulic codes using the Main Steam Line Break of a PWR as an exercise. There have been three phases in the benchmark: The separate models have been tested independently in the first two phases and the whole coupled code in the third phase. Thus the participants have had possibility to verify their models before focusing on the effects of the coupling methodologies. Statistical methodology has been applied in the comparisons because the benchmark is based on code-to-code comparisons. The next benchmark will be a BWR turbine trip. It is based on measured plant data and will therefore be very valuable.

The paper by **Nakajima et al.,** ***Three Dimensional Analysis of RIA in PWR and BWR***, deals with the modeling of high burnup effects during the typical Reactivity Initiated Accidents, control rod ejection accident in PWR and control rod drop in BWR. In recent experiments with high burnup fuel, there have been occurrences of fuel failures at lower deposited energy levels than had previously been assumed. Therefore, the licensing criteria of RIA in Japan have been revised to be burnup dependent. New detailed three-dimensional methods were used to evaluate the distribution of the enthalpy in the core and to understand the realistic fuel behaviour during the accidents. The fulfillment of criteria was checked with a best estimate code using conservative assumptions and carrying out sensitivity studies.

The paper by **March-Leuba and Verdu Martin,** ***Role of BE Codes in BWR Technology*** emphasizes that Best Estimate methods have always been needed in BWR calculations, especially in stability studies. Also the modeling of the recirculation loop dynamics plays an essential role in the calculations and it has always been included in the BWR models, there is no new need for coupling the neutronic and thermal hydraulic models as in PWR calculations. A review of the codes used is given. It is shown that accurate radial nodalization is needed in order to obtain reliable results because the stability depends non-linearly on local effects: the reactivity feedback effect has power-square weighting, core inlet flow distribution is important, the increase of instability of the high power channels cannot be compensated by decreased instability of the low power channels. Using the high-speed computers of today, predictive calculations of BWR stability can be done with the advanced best estimate codes.

The paper by **Tricot and Menant,** ***Coupled Neutronic and Thermal Hydraulic Modelisation Applied to the Fast Injection of a Slug of Deborated Water into the Core,*** deals with the important accident scenario of achieving criticality due to the heterogeneous dilution of boron in PWRs. In order to analyze this type of accidents detailed models are needed both for the three-dimensional aspects of the transfer of a deborated water slug into the core as well as for three-dimensional neutronics in the core. Details of the thermal hydraulic modeling are given. The main dilution sequences are explained and actions and improvements to avoid them are reported. Probabilities before and after the modifications are given. Due to the risk of reaching prompt criticality, IPSN considered that the subcriticality of the core should be increased from

1000 pcm to 2000 pcm in the shutdown state in order to avoid any return to criticality with up to 3 m^3 slug in the cold leg. However, EdF has also been encouraged to continue the improvement on the mixing calculation tools, including international comparison.

The paper by **Sievers et al.,** ***Thermal-Hydraulic Aspects of the International Comparative Assessment Study on Reactor Pressure Vessels under PTS loading (RPV PTS ICAS),*** compares Pressurized-Thermal-Shock analyses using different types of codes to calculate the thermal hydraulic mixing. The fracture mechanics tasks of the study are not discussed here. The partly significant scatter of results was largely understood being brought up due to differences in the analytical approaches used by the participants: correlation based engineering methods, system code, and three-dimensional computational fluid dynamics (CFD) codes. There were especially difficulties in recognizing the flow regime at the water-stripe discharge in the downcomer. Coarse-grid and parallel-channel techniques are not sufficient to provide local temperatures. Engineering models have difficulties in the transferability of results from tests to real plant applications. CFD codes have to be further developed in modeling of turbulence and multi-phase flows.

The paper by **Aniel-Buchheit et al.,** ***Calculation of a Reactivity Initiated Accident with a 3D, Cell by Cell Method: Application of the SAPHYR System to a Rod Ejection Accident in TMI1***, deals with sensitivity studies on the geometrical description accuracy, on the type of physical phenomena modeled, and on the values of key physical parameters applied. Significant differences in three-dimensional neutronic calculation results of a control Rod Ejection Accident (REA) are shown due to homogeneous or heterogeneous cell-by-cell assembly modeling. Different types of accidents should be studied to show the effects of hydraulic modeling or axial discretization because there are no large changes in these during a REA from zero power initial conditions. One transient is not sufficient to evaluate the influence of all parameters, but an “envelope transient” should be searched for each parameter.

The paper by **Bang et al.,** ***Recent Applications of Thermal-Hydraulic Code in KINS***, overviews the recent applications of RELAP5 in plant analyses for safety regulation and safety assessment evaluation. In Korea a peer-review has been carried out on the code assessment database by performing various integral and separate effect test calculations to see the importance of the physical phenomena involved. The basis, main calculation results and major findings are given of four typical thermal hydraulic accident cases of real plants. The needs for future applications are summarized including, e.g., three-dimensional hydrodynamic capability and coupling with three-dimensional neutronics.

TECHNICAL SESSION 3B: Needs for Integral Plant Transients and Accident Analysis, Known Limitations.

Session Chairmen: V. Teschendorff, H. Ninokata

The eight papers in this session covered a large variety of applications. The first four of them dealt with coupled analysis of 3D core neutronics and system thermal-hydraulics. Transients in VVER type reactors were analysed using different code combinations. NUPEC’s code system IMPACT comprises codes for DNB and CHF predictions, simulation of severe accident and other phenomena. Two papers from the industry enriched the session: A Canadian paper presented a systematic approach to formally qualify a code system, and a paper from Spain highlighted the benefits of integral plant calculations for a utility. One of the few papers dedicated to neutron kinetics reflected on UO_2 -based neutronics when applied to MOX-fuelled cores.

Summary of papers

Validation of Coupled Codes for VVERs by Analysis of Plant Transients
(S. Mittag, S. Kliem, F. P. Weiss, FZR; R. Kyrki-Rajamäki, A.Hämäläinen, VTT; S. Langenbuch, GRS; S. Danilin, KI; J. Hadek, NRI; G. Hegyi, KFKI; A. Kuchin, STCNRS; D. Panayotov, INRNE)

Different 3D neutron kinetics codes for VVER type reactors coupled to the thermal-hydraulic system codes have been compared against real plant transient data from two NPPs: Balakovo - 4 (VVER -1000) and Loviisa-1 (VVER-440). Participants from 7 countries participated in this EU-sponsored exercise. In both transients, core power and coolant circuit behaviour were closely coupled and affected by control rod movement. Generally, good agreement was achieved for the parameters compared. Some deficiencies and limitations were observed: In order to achieve accurate results at all core positions, the thermal-hydraulics of all fuel assemblies should be modeled individually; this still requires too much computing time when all these core channels are coupled to the coolant system. Future calculations could substantially benefit from a 3D representation of the lower and upper plena of the pressure vessel and of parts of the horizontal steam generators. Comparisons with real NPP transients are further limited by uncertainties in the measurements, e.g. position of control rods.

Advanced Analysis of Steam Line Break with the Codes HEXTRAN and SMABRE for Loviisa NPP
(A. Hämäläinen, T. Vanttola, VTT; P. Siltanen, FORTUM ENGINEERING)

For Loviisa (VVER-440) a postulated steam line break was analysed with the HEXTRAN-SMABRE code. Calculations show that increased break size doesn't necessarily lead to cooler water at core inlet. Only a mild return to power after scram is to be expected under worst conditions. Due to the asymmetric phenomena, proper analysis of the steam line break requires a 3D core calculation. Generally, the coupling of a 1D thermal-hydraulic system code with a 3D core kinetic code is sufficient for a realistic analysis of this case. Complicated processes such as asymmetric power generation, mixing in the reactor vessel, and various protection and control actions contribute to the scenario. The calculations with 3D kinetics and detailed plant model facilitate realistic plant safety calculations.

Development, Validation and Application of Tools and Methods for Deterministic Safety Analysis of RIA and ATWS Events in VVER-440 Type Reactors
(A.Keresztúri, G. Hegyi, M. Telbisz, I. Trosztel, KFKI-AEKI)

The code system of AEKI for reactor physics and accident analysis for VVER-440 was presented. KIKO-3D is a reactor dynamics code for VVER type reactors. It was validated against VVER-440 specific benchmark problems. ATHLET is a thermal-hydraulic system code for a wide range of transients and accidents. Two transients for VVER-440 reactors were calculated using the coupled KIKO3D-ATHLET code system for an asymmetric steam line break and the SMATRA code for an ATWS. For both cases, it was shown that the time dependent behaviour of core and coolant system could be analysed with the detail required for the safety assessment and that the acceptance criteria were fulfilled.

RELAP5-PANTHER Coupled Code Transient Analysis
(B.J.Holmes, G.R.Kimber, J.N.Lillington; AEA Technology; M.R.Parkes, British Energy Generation; R.Page, NNC)

A coupled RELAP/PANTHER model has been established for Sizewell-B, including a full 4-loop representation of the cooling system. The complete model has been assessed against data from two plant transients: a grid frequency error injection test at minimum stable generation power, and a single turbine trip event from full power. Good overall agreement with the plant data was achieved for both cases. The calculations demonstrated the advantages of using coupled T/H and neutronic codes for plant transient analysis. The importance of proper data interface management was underlined. Further work on balance-of-plant models, such as turbine and condenser, was recommended.

Overview of the Simulation System "IMPACT" for Analysis of Nuclear Power Plant Thermal-Hydraulics and Severe Accidents
(T. Ikeda, M. Naitoh, H. Ujita, N. Sato, T. Iwashita, T. Morii, K. Vierow, S. Nagata, NUPEC)

Outlines of three constituents, CAPE, FLAVOR and SAMSON, of the IMPACT code system for simulating DNB or CHF, fluid-structure interactions, and severe core damage phenomenology were presented. The performance of each code package was shown and the usefulness was emphasised. Typically, the accuracy of predicting DNB/CHF was of the order of 10%/5%. The FLAVOR code was able to reproduce the first instability region of the in-line oscillation due to symmetric vortex shedding. Development of a prototype code for severe accident analysis is still an ongoing effort. Several verification studies and integrated analysis showed its capability and applicability.

Qualification Programs for Computer Codes Used for Safety & Transient Analysis in Canada
(J. Pascoe, J.C. Luxat, OPG; B. McDonald, D. J.Richards, B. Rouben, AECL)

Systematic methodology and practices in Canadian nuclear industries were reported for formal validation and verification to assure a very low likelihood of significant errors and unquantifiable uncertainties in the computer codes, and to provide a documentation base that establishes software quality and to demonstrate to a regulator that the applicable requirements are met. It was the objective of this software qualification program to formally qualify the safety analysis codes for their intended applications. The program encompasses the majority of technical disciplines relevant to CANDU reactor safety analysis; outlines of the software Q/A processes, documentation and a sample of the practice were reported.

Benefits of Using Integral Plant Models in Utilities Availability and Safety Issues
(F. Reventós, ANACNV/UPC; C.Llopis, UPC)

Utilities are using thermal-hydraulic models of their own plants for a variety of purposes related not only to safety issues but also to availability and operational questions. The usefulness of integral plant models was emphasised in connection to validation of training simulators, verification of emergency procedures, PSA, and licensing purposes. Coupling of realistic control block models with a system transient analysis code would allow to improve substantially the final results not only from the operation but also safety point of view. The application of Best-Estimate analysis for licensing purposes is still limited by the fact that it has to be supplemented by an uncertainty study or a conservative calculation following an approved methodology.

Neutronics Methods for MOX Fuel Analysis in Light Water Reactors
(T. J. Downar, Purdue University; R.Y. Lee, US NRC)

A summary of limitations was presented when UO_2 **-** based neutronics were applied to a MOX fuelled core. Several modifications to overcome the deficiencies were proposed in modern nodal methods and examined for improving the accuracy of MOX fuel analysis. While these modifications have recovered much of the accuracy lost by the presence of MOX fuel, it was suggested that further benchmark analysis and assessment are necessary. The work presented in the paper addressed primarily steady-state neutronics; it was pointed out, however, that additional consideration is required for transient neutronics analysis of MOX fuelled cores.

OPEN TECHNICAL DISCUSSION SESSIONS 3A & 3B

Chair: V. Teschendorff
Panelists: Prof. J.M.Aragones, Prof. H. Ninokata, R. Kyrki-Rajamäki

With 15 papers altogether, Technical Session 3 with its two sub-sessions covered a broad spectrum of investigations and applications. Coupling of thermal-hydraulics with other processes was a central theme of many papers in this session.

Although neutron kinetics was the dominant coupling topic, other processes important for reactor safety were dealt with as well: Analysis of components and structures is making use of more accurate T/H boundary conditions, e.g. for the investigations of pressurised thermal shock (PTS) or of flow induced vibrations. Coupling of T/H with balance-of-plant (BOP) models for plant transient analysis or with core melt and fission product behaviour models for severe accident analysis have opened a wider field of application to thermal-hydraulic codes compared to their original purpose of LOCA and transient analysis.
The presentations covered the reactor types PWR, BWR, and VVER. The needs for further code development and modeling improvements were derived from these. Different from the Annapolis workshop, advanced reactor designs were not addressed.
The outstanding feature of this Session was the close connection of the papers to practical applications and real safety questions, among them boron dilution transients, transients starting from shutdown states, and IRAs with impact on high burn-up fuel. Questions of code maintenance and quality assurance were addressed as well. Coupled analysis of 3D neutron kinetics, detailed core thermal-hydraulics and plant system behaviour has become available for practical applications. This is a clear evidence of the maturity of the codes and the progress made in recent years.

Experiments were not presented by dedicated papers, since the workshop was on codes. They played a role in several papers, however, in the context of code validation. For neutronic calculations and coupled T/H neutronic calculations, benchmark exercises have shown once again to be an adequate means of checking on the accuracy of predictions. However, authors and participants in the discussion were well aware of the still remaining uncertainties in predictions, especially for coupled processes. The importance of real plant transients for the validation of coupled codes was stressed.

With this enlarged field of applications, limitations in the present codes have become visible and needs for further development have been identified.

Consolidated Achievements (Relevance H,M,L)

Neutronic codes using diffusion approximation with two energy groups are able to calculate transient behaviours of core power for PWR, BWR, and VVER reactors with appropriate boundary conditions. The core is discretized in 3D, with mesh sizes corresponding to one fuel element. Benchmark calculations provide a basis for estimations of the accuracy of these codes. (M)

Thermal-hydraulic codes are able to simulate the cooling system behaviour in transients and accident, using 1D elements for most of the components. 2D and 3D elements with coarse noding are applicable for the pressure vessel. Non-condensable gases and boron transport can be simulated. For complete plant simulation, the codes have various tools to model BOPs. (H)

For simulation of transients that involve strong interaction of core power and coolant system behaviour the 3D neutron kinetic codes are coupled to T/H system codes. Different mapping techniques are used for relating several fuel elements to hydraulic core channels. Validation of coupled code systems is based on benchmark calculations and a limited number of actual plant transients data. In most cases, a certain degree of conservatism is provided. Transients, design basis accidents, and ATWS can be calculated in a reasonable time. Parametric and sensitivity studies are possible in practice. (M)

Industries, i.e. utilities and vendors, are applying complete code packages with validate data sets for their plants, with emphasis on quality assurance and documentation. (H)

Current limitations.

The available neutronic codes are generally based on the diffusion approximation with two energy groups. Each fuel assembly is lumped, neglecting void distribution inside the bundle. This becomes a relevant limitation for very heterogeneous core loadings, e.g. for mixed cores with UO_2 and MOX fuel elements. (M)

The thermal-hydraulic system codes have only a limited 3D capability in the vessel. For practical coupled calculations, parallel channels are often used that require arbitrary mapping of corresponding fuel elements to hydraulic channels. (H)

For mixing problems, relevant for PTS or boron dilution transients, the system codes suffer from artificial diffusion caused by the first order space discretisation schemes. Front tracking models have not solved this problem. CFD codes with turbulence models are sometimes applied in sequential calculations to resolve the mixing process in specific reactor components. (M)

Development needs.

Reactor physics codes for accident analysis should allow for more than two energy groups. Neutron transport models should be evaluated regarding their potential applicability to transient analysis. More details should be provided by T/H to neutronics, e.g. subchannel void data. (M)

Thermal-hydraulic codes need enhanced 3D capabilities with mesh sizes corresponding to one fuel element. Less diffusive numerical schemes are needed to avoid artificial mixing. CFD codes with two-phase capabilities could solve the problem in a longer time perspective. Interfaces between CFD codes and plant system codes must be established. (M)

Validation of coupled T/H and neutronic code requires that additional data from actual plant transients become available. (M)

Other comments.

Analysis of transient core behaviour should not be confined to coupling of neutron kinetics and thermal-hydraulics but should consider fuel rod behaviour as well for certain transients. (L)

Industry should be more involved in defining actual needs and in providing plant data. (M)

A methodology to maintain a certain degree of conservatism should be developed. (M)

TECHNICAL SESSION 4: ADVANCES IN NEXT GENERATION OF TH AND NEUTRONIC CODES

Session Chairmen: H. Staedtke, J.C. Luxat

In this section, six papers were presented, which covered a wide spectrum of code development and improvement activities, including coupling of Thermal-Hydraulic and Neutronic codes, improved code user interfaces, increased code modularity, automatic code uncertainty evaluation and advanced numerical techniques for transient two-phase flow.

The paper by **H. Asaka** et al gives an overview on the progress in the ***Coupling of the Thermal-Hydraulic Code TRAC with the 3-D Neutron Kinetics Code SKETCH-N***. The 3-D neutron kinetics model SKETCH-N has been implemented into the transient analysis codes J-TRAC and TRAC-BF1. The coupled codes have been verified using various benchmark test cases. Results with acceptable accuracy are reported for typical PWR applications such as reactivity-initiated accidents. The assessment of the TRAC-BF1/SKETCH code system is underway using Ringhals-1 stability benchmark cases. Further information is given in the paper, which relates to the experimental evaluation of nuclear and thermal-hydraulic coupled instabilities in BWRs as presently performed at JAERI.

The paper by **J. Mahaffy** deals with the development of an ***Exterior Communication Interface (EIC) for the USNRC Consolidated Code*** which permits a tight coupling of the basic thermal-hydraulic code (e.g. the *USNRC Consolidated Code)* with other codes or code -modules for 3-D neutron kinetics, containment behaviour or with specific models for the AP600 design. The coupling will be realized via defined computational flow and synchronization points and the dynamic configuration of data transfer procedures. The new code structure will further support future extensions for parallel computing. It is expected that the EIC will enlarge the field of application for the USNRC's consolidated reactor safety analysis code presently under development.

The paper by **K. Jones** gives an overview on the development of the ***Symbolic Nuclear Analysis Package*** (SNAP) presently under way at SCIENTEC as part of the NRC code consolidation programme. The SNAP consists of four major modules: the *front-end* including an input model editor including an expert nodalization assistant and a 3-D viewer, the *runtime module* to control and monitor the execution of the TH-code, a p*ost processor*-module including visual engineering data analyzer and plotting software, and a *database*-module for storing the input models, results of calculations and system configuration information. Although the SNAP will not necessarily improve the predictive capability of the underlying TH-codes, it will certainly allow for a more efficient and consistent usage of the code and will possibly reduce the potential for user-related errors.

The paper by **I. Toumi** et al summarizes various activities presently performed at CEA (France), JRC-Ispra (EC), GRS-Garching (Germany) and EdF (France) on ***Advanced Numerical Methods for Transient Two-Phase Flow***. Common to all these methods is the use of first or second order characteristic based up-wind schemes, which principally require a hyperbolic nature of the convection part of the governing flow equations. These techniques, originally developed for single-phase gas dynamics applications, are characterized by low numerical diffusion/viscosity effects and as such allow for a high resolution of local flow phenomena including steep gradients or sharp discontinuities. Numerical examples shown indicate that the transfer of these techniques to two-phase flow has reached a certain degree of maturity. Nevertheless, before implementing in a standard or new system code, more extensive testing of the methods might be necessary to demonstrate their efficiency and robustness for typical reactor safety conditions.

The paper by **F. Kunz** et al relates to the development and testing of on ***An Automated Code Assessment Program (ACAP) for Determining System Code*** for the comparison of reactor system code results and experimental data resulting from a large number of batch code executions. It includes a collection of various measures for the evaluation of the data quality and procedures to produce graphical outputs for selected parameters and time windows. The paper summarizes the structure and properties of the ACP and its relevance for the TH-code development and assessment programme.

The paper by **W. Luther** on ***Highly Stable Time Integration Applying the Methods of Lines to Thermal-Hydraulic Models*** refers to the numerical method used in the German ATHLET code which somehow differs from other standard techniques as used for example in the RELAP5 or CATHARE codes. After space discretization, the resulting governing equations are interpreted as a finite difference form of a system of Ordinary Differential Equations, which is integrated by a general-purpose ODE solver. The author highlights the difficulties arising from the stiffness of the system of equations, the limited differentiability and the presence of discontinuities in the constitutive relations. Various ODF solvers are compared, indicating improved stability characteristics for fourth order accurate Runge-Kutta methods.

TECHNICAL SESSION 5: FUTURE TRENDS IN PHYSICAL MODELING FOR NEXT GENERATION CODES

Session Chairmen: Prof. G. Yadigaroglu, D. Bestion

Physical modeling cannot be treated in isolation from other aspects of code development and this is evident in the papers presented. In general, one remarks that some of the most important - but not all - physical model improvements that were identified as necessary in the 1996 Annapolis meeting are addressed. Six papers were presented in this session.

The paper by **Ishii et al.,** ***Interfacial Area Transport: Data and Models***, deals with an issue that was identified as important in Annapolis. This issue was addressed since on both sides of the Atlantic, the US NRC, as well as French organisations, initiated analytical and experimental programs, hopefully leading to improved modeling in the codes. The paper by Ishii et al. summarises the progress made in the US to replace the current, flow-regime dependent, interfacial area correlations with an interfacial area transport equation that could lead to dynamic flow regime transitions. This new approach produces continuous changes of the interfacial area and should eliminate the artificial, abrupt changes in flow regime and the related parameters, such as interfacial drag, produced by the present-generation codes using static flow regime maps. The French work in this area is briefly mentioned in the paper by Morel et al., also presented in this session; references on this important subject covering the basic aspects of the work can be found in that paper.
The Ishii et al. paper presents the first efforts to develop the interfacial area source terms based on mechanistic bubble interaction models; these were adjusted using data from experiments conducted at Purdue University and the University of Wisconsin at Madison. Preliminary work on the incorporation of an interfacial area transport equation in the US NRC consolidated code under development is also reported.

The paper by **Fry and Lillington,** ***Validation of the CFX-4 Code for PWR Fault Analysis***, deals with another issue considered as important in Annapolis, namely the future use of CFD codes for safety analysis. Indeed, current thermal-hydraulic system codes have limitations in modeling certain transients where turbulent mixing phenomena are important.
The paper presents the current status of the AEA Technology work validating CFX-4 as a tool for modeling fluid transport and mixing in reactor coolant systems. CFX-4 code predictions have been carried out against pump start-up data from 1/5 scaled experiments dealing with the transport of a boron depleted slug of fluid from the cold leg to the vessel; time scales and concentrations were well predicted. Predictions of thermal mixing have also been compared to Sizewell-B plant data from the Emergency Boration System commissioning tests. The code successfully replicated the broad features but failed to reproduce an observed swirl.

The paper by **Anghaie et al.,** ***Advanced Thermal Hydraulic Modeling of Two-Phase Flow and Heat Transfer with Phase Change***, deals with a Computational Fluid Dynamic (CFD) model developed to simulate LWR transients such as reflux two-phase flow and heat transfer with phase change. It is claimed that the model combines the high-resolution capability of state-of-the-art CFD methods with a novel approach that allows the tracking and delineation of the dynamic interfacial water-steam boundary; a number of additional advantages of the methods proposed are presented. An “internal energy fraction,” which is a parameter with both local and instantaneous value, plays a pivotal role in tracking the liquid-vapour interfacial boundary. This model is used to analyse a simple case of reflux condensation.

The paper by **Ninokata et al.,** ***Development of the NASCA Code for Predicting Transient BT Phenomena in BWR Rod Bundles***, describes state-of-the-art work in progress for improved analysis of transients in BWR rod bundles, including boiling transition (BT) and post BT phenomena. It is assumed that under BWR conditions, BT can be explained by liquid film dryout. Consequently a three-field model considering liquid films, entrained drops and the vapour is used. Major physical processes of significance are: two-phase flow turbulent mixing and void drift that affect the void distribution in a bundle; droplet entrainment and deposition phenomena, including spacer effects; rewetting after BT. Computational results are compared to experimental data, including BT experiments in 3x3 and 4x4 rod bundles. Interesting results showing transient dryout and rewetting were shown.

The paper by **Morel et al.,** ***From the Direct Numerical Simulation to Averaged Two-Fluid Models. How Different Types of Models Can Contribute to the Next Generation of Codes?,*** describes the ECUME initiative launched in 1998 by the CEA and EDF, leading to a co-coordinated strategy for the development of the next generation of two-phase thermal-hydraulic codes in France. A classification of codes in: system, component, CFD, DNS (Direct Numerical Simulation) and LES (Large Eddy Simulation) categories is made. The present component codes for core and steam generators and the system code CATHARE use either the 1-D, two-fluid model, or 3-D models in a porous-medium approach. Advanced models are being developed including transport for the volumetric interfacial area, turbulence modeling, and multi-field capabilities in 1-D, and in 3-D for both porous and open media. DNS techniques (that include, in this classification, Volume of Fluid or equivalent methods and front tracking) are also developed to investigate certain small-scale local phenomena and to be used as “numerical experiments” to produce closure relationships for averaged models.

The paper by **Lee and Chung,** ***MARS Development Program and Progress***, describes the systematic and extensive R&D program started in 1997 in Korea to develop the multi-dimensional, multi-purpose system analysis code MARS for the realistic thermal-hydraulic system analysis of LWR transients. MARS 1.4 was developed first as a unified code from RELAP5 and COBRA-TF. Subsequently, MARS 2.x is being developed as a multi-purpose system-analysis code with coupled multidimensional thermal-hydraulics and 3-D core kinetics, CHF and containment analysis capabilities. New features of the code, code modernization and restructuring, code assessment, and code coupling are described. The paper also briefly introduces companion KAERI experimental activities.

OPEN TECHNICAL DISCUSSION SESSIONS 4 & 5

Chair: Prof. G. Yadigaroglu
Panelists: H. Staedtke, J.C. Luxat, D. Bestion

Strictly speaking, only two papers were directly related to the improvement or to the development of advanced numerical methods for transient two-phase flow: the paper of W. Luther on *Highly Stable Time Integration Using the Methods of Lines* and the paper of I. Toumi on *Advances in Numerical Schemes for Two-Phase 3-D Thermal-Hydraulics.* This is a little misleading, however, because there has been progress over the last five or six years in the field of Computational Fluid Dynamics and flow simulation, which is of relevance also for Nuclear Thermal-Hydraulics.

It is of interest to recall first the main recommendations made for improvement or development needs at the Annapolis Meeting (adapted from a transparency presented by M. Réocreux):

Modeling areas:
- multi-field approaches (more than two fields)
- interfacial area transport approach
- multi-dimensional flow:
  - flow regimes need to be defined
  - interfacial coupling terms, need of experimental data with improved instrumentation
  - modeling of turbulent diffusion
- low pressure, low flow conditions: codes need to be tested
- performance of the existing models in the presence of non-condensable gases

Numerical Methods
- implementation of different numerical schemes for different phases of a transient
- numerical schemes for 3-dimensional flow
- need for low diffusive methods
- handling of wide range of problem time/length scales
- tracking of steep parameter gradients
- robustness
- high level of modularity
- implicit coupling of codes or code modules

It was proposed that further discussion consider the needs defined in Annapolis.

Synthesis of Technical Sessions 4 & 5 and corresponding discussion session:

RELEVANCE DESCRIPTION (H,M,L)

<u>Coupling of T/H and neutronics codes</u>

Consolidated achievements.
Significant advances have occurred since the 1996 Annapolis workshop, as reflected in the increased number of papers that dealt with aspects of coupling. Notable amongst these was the work on an *Exterior Communications Interface* for the US NRC Consolidated Code project, which provides a generalised interface for tight coupling between any combination of parallel executing codes, or processes. (H)

Current limitations.

Lack of a generalised protocol to guide interface implementation. (H)

Development needs.
Development of an interfacing protocol standard to consolidate international efforts. (M)

<u>Numerical schemes</u>

Consolidated achievements.
Steady progress has occurred in the development of improved numerical schemes that will enhance robustness of code solution capability. The focus of the work is specific to the particular codes that currently exist (e.g. ATHLET, CATHARE) or are being developed. Significant progress has been achieved for the development of a new class of high-resolution numerics, which could form the basis for more detailed 3-D code component modules. (M)

Current limitations.
Numerical methods presently used in TH system codes consolidate substantial improvements in terms of robustness; future improvements are expected to come from the continuous increase of computer power. (M)

Development needs.
No clear development needs were articulated at this meeting, but the need to proceed with extensive benchmarking of new numerical schemes and comparison with existing standard techniques was stated. (M)

<u>User Interface</u>

Consolidated achievements.
Major advances have occurred since the 1996 Annapolis workshop to develop user support tools to assist both the front-end input deck preparation and the verification and user run-time interaction. In particular, the Symbolic Nuclear Analysis Package (SNAP) developed under the US NRC Consolidated Code project provides functionality that meets the requirements articulated at the Annapolis workshop. (H)

A related achievement is the support tool for code validation, the Automated Code Assessment Program (ACAP), also developed under the US NRC Consolidated Code project. Use of the tool in the interactive mode can assist the user in quantifying the initial validation exercise results and, in the batch mode, it can provide a regression testing capability for any subsequent revalidation. (H)

Current limitations.

Not all developers are giving sufficient attention to this area – it appears to be viewed by some as less interesting work and hence, lower in priority. Schedule-driven projects may be the only way to ensure adequate attention. (H)

Development needs.

Leverage of resources through joint efforts and development of industry standard products should be pursued. (H)

<u>Interfacial Area Transport</u>

Consolidated achievements.
An application of the approach to the simulation of bubbly-slug flows has been presented, and it has been demonstrated that this approach has good potential for overcoming some limitations of current two-fluid models. (H)

Current limitations.
The approach has not yet been developed for all flow regimes and flow regime transitions. (H)

Development needs.
Extend to all flow regimes and flow regime transitions. Implementation in 3-D models is required for two-phase CFD codes. (H)

<u>Multi-field Models</u>

Consolidated achievements.
A 3-field model has demonstrated the capability to handle complex sub-channel fluid phenomena that influence boiling transition heat transfer and rewetting. The phenomena involved include turbulent mixing, droplet entrainment and deposition and local turbulence promoting effects of spacer grids. (H)

Current limitations.
Only 3-field models for annular-mist flow are well advanced in the nuclear thermal-hydraulic field. (M)

Development needs.
Definition of the approach combining multi-field capability with interfacial area transport for all flow regimes is required. (H)

CFD Codes

Consolidated achievements.
Applications have been performed for some single phase transient flow cases and the indications are that CFD will be a very useful tool for certain situations, e.g., when single phase turbulent mixing governs the transient behaviour. (H)

Current limitations.
Applied so far to single-phase problems and, to a lesser extent, to some two-phase dispersed flows. Limitations exist in the choice of turbulence models, and the widely used k-epsilon model has deficiencies. The choice of nodalisation remains difficult. (H)

Development needs.
Need for more detailed user guidelines for problem-related grid generation and selection of specific code options for turbulence models and numerical details. (H)

Advanced Two-Phase Numerical Simulation Techniques

Consolidated achievements.
Rapid progress has occurred in the past few years. These techniques can simulate microscopic phenomena and can provide interface-tracking capability. They hold the potential for use as tools to complement experiments in the development

Current limitations.
The development and application of such techniques are still in their infancy.

Development needs.
Develop for use as a support tool for modeling closure terms and for dealing with special problems. (M)

SESSION: 6a UNCERTAINTY ANALYSIS, LEVEL OF CONFIDENCE

CHAIR: Prof. F. D’Auria, M. Hrehor

The Session 6a deals with the uncertainty evaluation (papers **6a1** and **6a2**) and with quality assurance procedures needed to achieve suitable reliability in the results (paper **6a3**). The uncertainty evaluation must be seen as an indispensable supplement of any Best-Estimate code calculation. This derives from unavoidable approximations that are embedded into the codes and in the procedures of codes application to the cases of interest. In facts, approximations are present in the physical models, in the numerical solution schemes and in setting up the interface between the code and the system to be simulated (i.e. nodalisation development process). Approximations in the nodalisation development remain notwithstanding the availability and the use of the qualification procedures discussed in one of the papers. The uncertainty evaluation concretizes into the derivation of error bands that bound any typical time trend predicted by the code or, more simply, the value of any calculated parameter (e.g. the Peak Cladding Temperature). Continuous error or uncertainty bands are derived in the first situation that is also the situation addressed by the two concerned papers in the session.

SUMMARY OF PAPERS

Uncertainty Evaluation of Code Results, Application to SBLOCA (H. Glaeser, GRS)

The quality demonstration of the GRS uncertainty method and the application of the same method to a transient calculated for a NPP are part of the paper. The quality demonstration is achieved by utilizing the experimental data considered in the OECD/CSNI UMS (Uncertainty Methods Study). The basic advantages of the method are:

- the independence of the number of calculations needed for uncertainty evaluation from the number and the type of input uncertainties and
- the possibility to distinguish individual contributions to the overall calculated error or uncertainty.

It is proved that the method can be used for licensing applications and, in the general case, for the prediction of NPP related scenarios.

The Capability of Internal Assessment of Uncertainty (F. D´Auria, Univ. Pisa, P. Marsili, ANPA)

The Internal Assessment of Uncertainty (IAU) constitutes a capability that was requested for system codes in the OECD/CSNI Meeting held in Annapolis in 1996. The advantages of IAU over an uncertainty method are:

- results of uncertainty method applications are not affected by the user of the uncertainty method;
- almost no resources are needed for the application of the uncertainty method.

This implies that all the necessary engineering choices are embedded into the method and that no panel of experts is needed for the application of the method. The paper shows that the IAU capability has been achieved by combining the Relap5 code (US NRC version) and the UMAE uncertainty method. The CIAU (Code with capability of IAU) has been proposed. The basis of the CIAU is discussed in the paper, as well as significant results obtained from its application.

Qualifying, Validating and Documenting a Thermalhydraulic Code Input Deck (C. Pretel, L. Batet, A. Cuadra, A. Machado, G. de San José, I. Sol, F. Reventós, UPC).

The work documented in the paper starts from the consideration of the huge effect that the code user may have upon the results of predictions made by Best Estimate codes. The need for a consistent quality assurance procedure is identified. It is recognized in the paper that 'Documenting a model means much more than the initial task of justifying or describing geometry, controls and protections, kinetics, ... It needs to be a dynamic process regulated in some way and submitted to the standard quality control process of the utilities'.

On these bases, an integrated procedure is proposed that adequately brings to a reduction of the user effect. The procedure basically covers the most important logical steps that are pursued when developing an input deck. The consideration of a procedure similar to what proposed must be a pre-requisite for applying any of the uncertainty methods or approaches discussed in the previous two papers. Any code application (see also below) should benefit from such a procedure.

SESSION: 6b SIMULATORS AND FAST RUNNING CODES

CHAIR: Prof. F. D'Auria, M. Hrehor.

The Session 6b deals with the demonstration of the possibility of using system codes as fast running codes in the areas of NPP simulators (paper **6b1**), of Probabilistic Safety Assessment (paper **6b3**) and of licensing (paper **6b2**).

The availability of more and more sophisticated computers, as well as of procedures for the effective exploitation of the capabilities of the codes and of the computers themselves, broadens the application domain of the codes. The availability of frozen/qualified code versions and of sophisticated graphical-users-interfaces is indispensable in this context and largely facilitates the learning process for the use of the codes. In this way, the Best Estimate codes become accessible to a wider number of users, not necessarily thermalhydraulics specialists. PSA analyses, simulator applications and licensing studies, specifically performed on a real time basis, also constitute relevant reasons for justifying the development and the improvement/qualification of the system codes.

SUMMARY OF THE PAPERS

The SCAR Project: How a Best Estimate Code Can Be a Fast Running Code (J.M. Dumas, IPSN, F. Iffenecker, EDF, M. Farvacque, CEA)

This is the first of the three papers dealing with an important area of code applications. The concerned area is the use of a thermalhydraulic system code, Cathare 2, for developing a multipurpose simulator. A wide range R&D work is still in progress and involves the major actors in the nuclear technology in France, i.e. EdF, CEA, IPSN and Framatome. The benefit expected, over the alternative/simplified approaches pursued in the past, lies in the increase in the level of confidence for the output provided by the simulator. This is achieved by the use of the original code version and at the expense of a huge program of optimization of important numerical steps necessary for achieving the solution.

The huge increase in parallel computer capacity makes possible the achievement of the mentioned objective, i.e. the use of a complex thermalhydraulic code into the simulator. The achievement of real time simulation for any generic transient is expected for the year 2003.

LBLOCA Analyses with APROS to Improve Safety and Performance of Loviisa NPP (H.Plit, H.Kantee, H.Kontio, H.Tuomisto, Fortum Eng.)

This is the second of the three papers dealing with an important area of code applications. The concerned area is the Best Estimate system code application to the licensing process and to the safety evaluation of a nuclear power plant. Emphasis is given in this context to the Large Break LOCA evaluation. One of the significant results is the confirmation of the possibility of up-rating the core power of the Finnish Loviisa NPP. This must be seen as an actual way for the industry to get back the financial resources spent for the development and the qualification of the system codes.

A few remarkable aspects connected with the code validation, as well as code validation results, are discussed in the paper, making reference to the APROS code. The advantages of 'maintaining' different code versions with different complexities are stressed in the paper. The authors also give an idea of the complexity of the developed NPP nodalisation by mentioning that more than 40000 conduction heat transfer meshes are used in the core alone.

Role of Fast Running Codes and their Coupling with PSA Codes(J.M. Izquierdo, CSN, C. Queral, R. Herrero, UPM, J. Hortal, M. Sánchez, E. Meléndez, R. Muñoz, CSN)

This is the third of the three papers dealing with an important area of code applications. The concerned area is the integration of the Best Estimate system code and the Probabilistic Safety Assessment (PSA) analysis. The current status and some significant results of a pioneering research are discussed in the paper: a fast running thermalhydraulic code is coupled with a typical PSA computer tool. In this way it is possible to generate automatically event trees, thus adding an innovative level to the investigation on the safety of NPP. One of the needs for engineering judgement in the PSA is avoided in this way. The availability of fast running, 'frozen' and robust system codes is a prerequisite for the full integration of the code within the proposed methodology.

Pilot applications of the methodology have been completed in relation to PWR, while BWR related studies are in progress.

OPEN TECHNICAL DISCUSSION SESSIONS 6a & 6b
CHAIR: Prof. F. D'Auria, M. Hrehor

Summary of the discussion

Additional questions to each of the six papers were provided during the open discussion and were answered by the authors. Two topics were addressed in the subsequent discussion, as foreseen in the scheduled program: a) Uncertainty and BE codes, b) Simulators and BE codes. The summary of the discussion has been split into two parts, achievements and recommendations related to the session topics and to the items a) and b) above.

Achievements.

The following are considered as achievements in the area:

- The uncertainty methods have advanced to a reasonable level and are ready for practical applications. This is partly demonstrated by the OECD/CSNI UMS study and by the subsequent activities carried out in the framework of the scientific community.
- The use of BE codes plus uncertainty methods is allowed by the regulatory authorities all over the worlds, though on a case-by-case basis. This was confirmed by representatives of the authorities of countries like USA, UK, France and Spain. Information in the same direction was provided in relation to other countries like Netherlands, Canada and Brazil.
- The usefulness of system codes was confirmed making reference to the following applications:

  - the area of simulators (French Scar Project);
  - the coupling between PSA and thermo-hydraulics (long term investigation at CSN, Spain);
  - the upgrading of plants (Fortum utility in Finland): benefits to the safety and the operation were emphasized.

Recommendations.

The following recommendations came out from the discussion:

- The approaches to the evaluation of uncertainty should be simplified. An example of how this can be done is given in the paper 6a2. This also constitutes an answer to one of the needs (the Internal Assessment of Uncertainty) expressed at the Annapolis OECD/CSNI Meeting of 1996. However, the simplification process should be aware of potential new problems.

- The industry (utilities, designers, fuel vendors) should take more benefit from the availability of these methods: benefits can especially come from setting up cooperation with regulatory authorities.

- The developers of uncertainty methods should be able to help in answering questions coming from different potential end-users of the methodologies.

Synthesis of SESSION: **6a UNCERTAINTY ANALYSIS, LEVEL OF CONFIDENCE**

RELEVANCE DESCRIPTION (H,M,L)

Consolidated achievements.

The uncertainty methodologies have reached a suitable level of development. They can be effectively utilised in the licensing process of existing reactors. (H)

The feasibility demonstration of Internal Assessment of Uncertainty (IAU) has been achieved. This constitutes a remarkable follow-up to the Annapolis Meeting. (H)

Quality assurance procedures when developing nodalisations are needed to limit the user effect upon Best Estimate system code predictions. Valuable examples of the structure of these procedures are available and have been discussed. (H)

Current limitations.

Common understanding about features and capabilities of uncertainty methods has not been reached. This is also true in relation to differences between uncertainty methods that are based upon the “propagation of code input uncertainties” and based upon the “propagation of code output errors”. In the former case propagation might occur through a physically imperfect model. In the latter case the propagation relies upon the quality of measured data. For both cases qualification proofs have been achieved. (M)

Limitations of uncertainty methods might be connected with limitations of the Best Estimate codes. However, it must be emphasised that uncertainty methods are designed to overcome current limitations of system codes. (M)

Development needs.

The complexity of the methodologies and the difficulty in proving analytically the correctness of the provided results, bring to the following proposal:

“Pioneering applications of uncertainty methods within regulatory processes should include the use of two independent methods or use of the same method by independent groups”. (M)

The IAU capability (see above) should be proved using uncertainty methods and codes different from those discussed in the session. (M)

Other comments.

Competences in system thermalhydraulics, including development/use of system codes and uncertainty methodologies should be kept. Coordinated efforts are needed among the major international organisations (OECD, CEC, IAEA, US NRC, DOE, etc.) (H)

Experiments are always necessary to confirm to prove main findings/achievements in system thermalhydraulics. (H)

Synthesis of the SESSION: 6b SIMULATORS AND FAST RUNNING CODES

RELEVANCE DESCRIPTION (H,M,S)

Consolidated achievements.

Use of Best Estimate system codes for optimising current NPP design and EOP (Emergency Operating Procedures). (H)

Use of Best Estimate system codes as basis for NPP simulators, suitable for operator training and visualization of complex system performances. (H)

Use of Best Estimate system codes in areas like PSA (Probabilistic Safety Assessment). (H)

Current limitations.

Large resources (several tens of man-years) are still needed to exploit all the capabilities of currently available system codes in the three areas above identified (category 1). This is especifically true for the area of coupling between thermalhydraulics and PSA. (M)

Development needs.

Experts from different domains of nuclear reactor safety should be more prone to exchange information and competences. This is valid for all relevant sectors having any connection with the system thermalhydraulics. (M)

EOP, specifically in case of low or very low probability accidents, can be widely optimised by using available tools. This is more valid in relation to Eastern designed and operated reactors. (H)

FINAL PARALLEL SESSION A: Coupled TH/Neutronic codes

Session Chair: Prof. J.M.Aragones
Co-chair: V. Teschendorff

Consolidated achievements.

All large system TH codes, and several 3D core TH codes, have been successfully coupled to state-of-the-art 3D neutronic-kinetic nodal codes, through general and modular interfaces. (H)

The NEA/NRC Benchmark on PWR MSLB transient has been successfully completed in the last 3 years, providing a demo of the computational feasibility and estimation of the accuracy of coupled TH/N 3D codes. (H)

Specifications of the NEA/NRC Benchmark on BWR Turbine trip transient have been issued.
(H)

The 3D neutronic codes that have been, or can be, coupled to the TH system and/or core codes are essentially the same used for cycle reload, design analysis, and licensing, and operation surveillance, providing the initial validation database that should be enlarged for best-estimate transient analysis. (M)

Current limitations.

The Neutronic/Thermal-hydraulic coupling is done explicitly or semi-explicitly, with time steps that are just input, heuristically derived or automatic. (M)

The fuel thermal properties and fuel-to-clad conductance used in TH codes are either too simple, without account for actual thermo-mechanical history, or lack the time dependent effects, needed for accurate Doppler feedback. (M)

The existing uncertainty analysis methodologies are limited to adjoint flux and nuclear data covariance methods or statistical C-M analysis for steady-state, nominal or HZP conditions. (M)

Decay heat uncertainties and realistic best-estimate methods are needed for some transients. (L)

Development needs.

Experimental qualification of coupled 3D neutronic/thermal-hydraulic codes with actual transient data of operating NPP. Transient data from experiments are only relevant if fuel and core geometries are close enough to NPP. (H)

Robust and automated coupled time - step control methods, for explicit and semi-implicit coupling including the switching on/off the N or TH code. Transient cross-section libraries, spanning the full parameter space. (M)

Uncertainty analysis for 3D kinetics and coupled N-TH methods.
Revision and experimental qualification of physical models (transport and spectral) of 3D nodal codes for high burnup and advanced fuels and reactors. (M)

Transient pin-by-pin and subchannel reconstruction for detailed time - dependent fuel rod thermo-mechanical analysis (off line). Feed back of fuel rod transient properties. (M)
User interface for QA consistent N/TH data input (L)
High performance computing and parallelization (L)

FINAL PARALLEL SESSION B: Use and applications of BE codes

Session Chair: J. Uhle
Co-chair: F. Reventos

Uncertainty Methodologies

New concept:

There are several uncertainty methodologies available for use with BE codes, and have been reviewed under the Uncertainty Methodology Study completed by the PWG2. A point of contention during the session was that each of these methods may yield a different answer for the range of uncertainty of a specific code for a specific calculation. This complication requires that the regulator and licensee must agree on what must be submitted and how it will be used to ensure that safety is maintained. It was suggested that the definition of BE methodologies reflect this range in uncertainty values, as it has not been included in previous descriptions:

BE methodologies are combinations of codes (without a severe bias to conservative values, i.e. as good as we can do), a qualified plant model and uncertainty methodology. The results of the code together with its corresponding uncertainty value allow the safety of the plant to be evaluated. Due to uncertainties in code models, input models, plant status and application uncertainty methodology, care should be exercised in BE analysis to demonstrate an acceptable level of confidence.

Achievements:

Uncertainty Methodology Study completed under the sponsorship of CSNI PWG2. Participants include: Germany, Italy, Spain, France, UK

Requirements:

Uncertainty methodologies must be simplified.

Uncertainty methodologies should be applied to coupled codes, such as kinetics codes. In the case of neutronics codes, a new approach must be developed to provide a measure of accuracy, since limited data exists. Some suggestions included comparing to Monte Carlo cases, although highly CPU intensive, benchmarking to other kinetics codes using different solution schemes (i.e., ANM vs. NEM), or comparison to transport codes.

Use of BE Codes

Achievements:

The uses of BE codes to date generally fall under three categories, including research oriented studies, design and operational support and one new application, PSA in Spain. Although limited, some licensee submittals have utilized BE methodologies internationally. These countries include Brazil, the Netherlands and Canada.

BE codes provided noticeable benefits to increase operation stability and/or gain margin.

The use of BE codes helps to improve the availability of existing power plants and can be considered a step forward in unifying safety and competitiveness goals.

<u>Quality Assurance</u>

Input Decks

Issue:

In any BE submittal, the code and uncertainty methodology are not more important than the pedigree of the input deck. Therefore, it was agreed that effort should be spent on qualifying input decks and their development cycle.

Achievements:

Prof. Pretel of Spain presented a useful tool to assist in the validation and documentation of input decks as an example of one such approach.

The SNAP GUI under development in the US also provides a means of controlling the development cycle of input decks.

<u>Code Validation</u>

Issue:

As most BE codes are not frozen, care must be taken to ensure that the fidelity of one DA case is not enhanced at the expense of another phenomenon.

Determining the adequacy of BE codes for a particular transient of a particular design involves comparing the code predictions to data and evaluating the fidelity. Subjectivity in the estimation of agreement can limit the validity of this approach.

Achievements:

The ACAP tool developed in the US can be used to automate this code comparison and generate numerical ranking of similitude between code prediction and data once a cognizant engineer studies the case and establishes the proper numerical techniques to use and performs data conditioning. Once this engineering judgement is applied, the process can be automated and repeated, with an aim of minimizing subjectivity and effort.

FINAL PARALLEL SESSION C: Future R&D in TH modeling and numerics

Chair: D. Bestion
Co-chair: H. Staedtke

Two topics are covered in this session in view of elaborating recommendations for the improvement of:
- Thermalhydraulic physical modeling
- Numerical schemes

TH PHYSICAL MODELING

Interfacial area modeling

Consolidated achievements:

The recommendation was made at the Annapolis workshop to develop new two-phase flow models including the transport of the interfacial area density. Achievements in this domain since Annapolis have confirmed the technical interest of this new approach.

Development needs/ Recommendations

Two main recommendations arose from the discussions:

- Investigations, that were limited to bubbly and slug flow regimes, should be extended to the other flow regimes and to all flow regime transitions for both 1-dimensional and multi-dimensional models.
- A comprehensive program of experiments required to elaborate the physical models related to interfacial area transport should be established, including adiabatic and diabatic flows with phase change.

Multi-field modeling

Consolidated achievements:

Recent results obtained with a three-field model for annular-dispersed flow have shown advanced capabilities for investigating and modeling local complex phenomena (entrainment-deposition, effects of spacer grids, turbulent mixing effects,…)

Development needs/ Recommendations

Two main recommendations arose from the discussions:

- Looking for advanced modeling based on a multi-field approach, a limited number of fields should be considered, since the experimental information available for qualifying the closure models is and will be rather limited.
- Annular-mist flows appear as an example where a multi-field model is most likely to provide significant progress compared with current two-fluid models.

Use of CFD codes in single phase flows

Consolidated achievements:

CFD codes have been applied rather successfully to some accidental transients related to mixing problems.

Development needs/ Recommendations

- It is recommended to establish more detailed user guidelines with regard to grid generation and the selection of specific code options for turbulence modeling and numerical details.
- Since more experience in the use of CFD codes is needed, one could take advantage of exchanging views with other industrial fields where CFD codes are more extensively used.
- More investigations regarding buoyancy effects using CFD codes are needed.

Use of CFD codes in two-phase flows

Looking for a finer space resolution of 3-D Two-phase flow models, advanced models in progress will become CFD type tools.

Development needs/ Recommendations

- It is recommended to put more effort into the modeling of turbulence for two-phase 3-D flow processes.

Use of two-phase DNS techniques

Consolidated achievements:

DNS techniques (such as Volume Of Fluid, Level Set, Front Tracking, Second Gradient,…) with interface tracking made significant progress during the past few years.

Development needs/ Recommendations

- It is recommended to use such methods complementary to experimental investigations as a support for the development of closure relations for averaged equation models, including interfacial area transport and multi-field models.

NUMERICAL SCHEMES

Consolidated achievements:

Following the recommendations made at the Annapolis meeting, considerable effort has been spent in several OECD member countries for the development of advanced, high-resolution low-diffusive numerical techniques, in particular with regard to multi-dimensional flow simulation. As was reported at the present workshop, encouraging progress has been achieved and there is a good indication that the new numerical schemes will reach the necessary degree of maturity within the near future. Important numerical features identified are related to handle detailed hydraulic networks and complex geometries including:

- Use of unstructured grids
- Automatic grid adaptation techniques

Development needs/ Recommendations

For the continuation of this ongoing effort, the highest priority should be given to:

- Low diffusive methods, at least second order accurate in space
- Implicit and/or higher order time integration

- Automatic build-in time step control based on convergence checks of iteration procedure and or accuracy estimation

In order to monitor the progress in developing new, advanced numerical methods and to check whether these techniques match the basic requirements as given above, it was recommended:

- to select a list of numerical benchmark test cases and to proceed to an extensive comparison of new schemes with the current ones.

<u>Code structure and interfaces:</u>

Consolidated achievements

Following recommendations made at the Annapolis meeting, some advances were reported at the Barcelona Workshop. Achievements concern four main topics:

- Modular structure: clearly defined interfaces between physical description and numerical methods are now commonly used
- Coupling of TH codes with other codes: coupling with neutronics using mainly explicit external coupling techniques were reported
- Parallelization: parallel versions of system codes with up to 8 processors were reported using either pvm or mpi.
- User interface and Pre-Post processing: Graphical User Interfaces are being developed including assistance in input preparation, online graphical display of results and graphical outputs.

Development needs/ Recommendations

The main recommendations for future developments are related to:

- Increased degree of implicitness in coupling TH codes with other codes
- Extension of parallel computing capabilities
- Completion of GUI developments

As a mid term activity, it is also recommended to enlarge the modularity of the code structure allowing in the future a more easy extension to increased modelling details (multi-field capabilities, interfacial area transport).

CONCLUSION OF THE MEETING by M. Livolant, Director of IPSN, France and Chairman of CSNI

During these days, I have followed with great interest the presentations, and discovered many directions in which improvements to the actual codes can be made, with some difficulty to appreciate what improvement will become a reality for the code users, and when that will happen.

In fact which are the needs for the code users. I cannot speak for all the types of code users, and I can express only the needs for the engineers of my institute. Our task is to verity the safety of the French Nuclear Power Plant, on the base of our own studies, or by controlling the utility safety reports and to prepare scenarios for accidents used to train the emergency crisis French system.

For that the main areas of the codes are for example:

* Probabilistic safety analysis with calculations as realistic as possible, in situations like:
  -Time delay for operator action.
  -Verification of automatic actions.
  -Physical conditions at the beginning of core uncovery.
  -Shut down situations.

* Studies with uncertainties treatment.

* Use of simulation system for accident scenarios, or procedures verification.

For all such studies, the actual generation of codes is very helpful, and gives a relatively reliable estimation of the safety margins, and of the evolution of accident sequences.

Nevertheless, they have some limitations due to the inherent simplifications of their physical models and numerical methods, like 1D instead of 3D, correlations established in static condition for transient situations, 2 fluids, approximate turbulence models.

One of the topics of the workshop is the coupling between thermalhydraulic and neutronic kinetics. Four our type of pressurized water reactors, the number of situations where this coupling appears necessary is limited.

- Steam pipe rupture.
- Reactivity insertion accidents following a control rod ejection or boron dilution problems.

I got the impression that this problem is already solved by the external coupling of existing codes - typically in France the chain: cathare-flica-cronos- with one question raised: is the external coupling sufficient in all the situations we have to consider?

Still in the position of end users for safety studies, we have other needs of thermalhydraulics studies.

1.- Thermalhydraulic in the containment in case of LOCA or severe accident, coupled with fission products transportation and hydrogen repartition.

For that we use at the present time lumped parameters codes, which have strong limitations of use and just begin to do 3d calculations with cfd codes for limited parts of the containment with lengthy calculation.

2.- Severe accidents for calculation of core degradation, late phase of corium in vessel progression, vessel bottom rupture, corium spreading and cooling.

There is a need for coupling with system codes to have a complete view of the situation of evolution during the accident. This is clearly in progress in France with CATHARE-ICARE codes.

3.- Thermalhydraulic for mechanical calculations. Typically, the understanding and prevention of some fatigue problems, which can go up to pipe ruptures, needs better calculations of hot-water/cold-water mixing with possible fluctuations and stratification effects in some parts of the circuits.

4.- Studies where thermalhydraulics are to be combined with combustion models.

So the need for improvements in thermalhydraulics is not limited to the treatment of the primary circuit in case of LOCA, and we can expect that even if each domain is specific by its physical situation, some generic methods are of common interest and, as I have heard during this meeting are of common interest also with non-nuclear application.

If I come back to the main topic of the conference and try to synthesize the situation, it appears that we have now a level of tools which allows for some confidence in Best Estimate calculations and are largely used by utilities and regulators and could even be more useful with some minor improvements and with a clearly established treatment of uncertainties.

In such a situation, and in a period where the available funds for research are reduced, one first option for the persons who have to decide on the best utilization of the research resources is to focus on improving the methods of utilization of the existing tools, and to limit codes improvements to what is possible in a short term period.

My Institute, the IPSN, has the responsibility to verify that the safety level is maintained, and, if possible, improved, in the French Nuclear Power Plants. For that objective, we estimate that the thermalhydraulic field is essential, and has the possibilities of improvements and all that justifies to keep an active research on a long-term basis.

Such an opinion is shared up to now by the other French main contributors in the field, typically CEA, EDF and FRAMATOME.

The main idea is to set up an organization for the post CATHARE situation, in order to support the continuation of the work on:

* The development of physical and numerical models.

* The development of instrumentation and experimental work specially oriented to the validation of physical improved models and after significant progress made.

* The development of a new generation code.

I have personally the opinion that such an operation should be open to international collaboration, especially in the European area. Some actions in that direction are already going on -and a European research program called EUROFASTNET has been settled, to deal with the more fundamental topics.

I hope that the other countries active in the fields will continue research in this field, in order to maintain a competition, which is a good stimulant for researchers. There may be some concerns for that when one sees the important reduction of funds for safety research in the last years and in some places the deficit of new young people coming into the field.

That type of concern is followed with great attention in the CSNI, to be sure that the capabilities are maintained at a sufficient level, and to initiate actions if necessary. A special group will be established at the level of the Committee and the Bureau, called the Program Review Group. They will be in charge so as to have an overview of the programs going on and the work of the working groups and to report to the Bureau and the Committee. Michel Réocreux will be a member of this group, and no doubt that he will follow with high attention how things are going in the thermalhydraulic field.

As you know, there has been a change in the structure of the CSNI groups. The very well known PWG-2 and PWG-4 are now unified in a group called Accident Analysis Group.

The PWG-2 was during a long time the heart of the international exchange in thermalhydraulics of the primary circuit. It organized many benchmarks and workshops, and defined a well-known validation matrix for the thermalhydraulic codes. We have to thank all those who made this action possible, and many of them are here.

We expect that the new group, with some new people inside, will do also a good work, in supporting the efforts made in the OECD countries to keep an active thermalhydraulic community despite the background of credit reductions everywhere.

I see at the occasion of the creation of this new group an opportunity to increase the collaboration between the various groups concerned with thermalhydraulic problems, like those encountered in the containment or in severe accidents. The question is not to mix everybody in the same group because the physical situations are different, but to stimulate exchanges in generic techniques, like CFD calculations for example.

I would not like to finish without expressing the thanks of everybody for our hosts in Barcelona, Mr. Pelayo and Prof. Alonso all their collaborators, and our Secretary Mr. Hrehor. They gave us the better conditions for a successful meeting, in this beautiful Barcelona town.

Thank you.

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