NUREG/CP-0158
NEA/CSNI/R(96)3

Proceedings of the OECD/CSNI Specialists Meeting on Boron Dilution Reactivity Transients

Held in State College, Pennsylvania, USA October 18-20, 1995

Organized by

OECD Nuclear Energy Agency

The Pennsylvania State University

Sponsored by
Office of Nuclear Regulatory Research
U.S. Nuclear Regulatory Commission

0/1
DF02

AVAILABILITY NOTICE

Availability of Reference Materials Cited in NRC Publications

Most documents cited in NRC publications will be available from one of the following sources:

1. The NRC Public Document Room, 2120 L Street, NW., Lower Level, Washington, DC 20555-0001

2. The Superintendent of Documents, U.S. Government Printing Office, P. O. Box 37082, Washington, DC 20402-9328

3. The National Technical Information Service, Springfield, VA 22161-0002

Although the listing that follows represents the majority of documents cited in NRC publications, it is not intended to be exhaustive.

Referenced documents available for inspection and copying for a fee from the NRC Public Document Room include NRC correspondence and internal NRC memoranda; NRC bulletins, circulars, information notices, inspection and investigation notices; licensee event reports; vendor reports and correspondence; Commission papers; and applicant and licensee documents and correspondence.

The following documents in the NUREG series are available for purchase from the Government Printing Office: formal NRC staff and contractor reports, NRC-sponsored conference proceedings, international agreement reports, grantee reports, and NRC booklets and brochures. Also available are regulatory guides, NRC regulations in the *Code of Federal Regulations*, and *Nuclear Regulatory Commission Issuances*.

Documents available from the National Technical Information Service include NUREG-series reports and technical reports prepared by other Federal agencies and reports prepared by the Atomic Energy Commission, forerunner agency to the Nuclear Regulatory Commission.

Documents available from public and special technical libraries include all open literature items, such as books, journal articles, and transactions. *Federal Register* notices, Federal and State legislation, and congressional reports can usually be obtained from these libraries.

Documents such as theses, dissertations, foreign reports and translations, and non-NRC conference proceedings are available for purchase from the organization sponsoring the publication cited.

Single copies of NRC draft reports are available free, to the extent of supply, upon written request to the Office of Administration, Distribution and Mail Services Section, U.S. Nuclear Regulatory Commission, Washington, DC 20555-0001.

Copies of industry codes and standards used in a substantive manner in the NRC regulatory process are maintained at the NRC Library, Two White Flint North, 11545 Rockville Pike, Rockville, MD 20852-2738, for use by the public. Codes and standards are usually copyrighted and may be purchased from the originating organization or, if they are American National Standards, from the American National Standards Institute, 1430 Broadway, New York, NY 10018-3308.

NUREG/CP-0158
NEA/CSNI/R(96)3

Proceedings of the OECD/CSNI Specialists Meeting on Boron Dilution Reactivity Transients

Held in
State College, Pennsylvania, USA
October 18-20, 1995

Manuscript Completed: March 1997
Date Published: June 1997

Organized by
OECD Nuclear Energy Agency
The Pennsylvania State University

D. Ebert, NRC Project Manager

Sponsored by
Division of Systems Technology
Office of Nuclear Regulatory Research
U.S. Nuclear Regulatory Commission
Washington, DC 20555-0001
NRC Job Code L1971

NUREG/CP-0158 has been reproduced
from the best available copy.

ABSTRACT

A CSNI Specialist Meeting on Boron Dilution Reactivity Transients was held in State College, Pennsylvania U.S.A., from October 18-20, 1995. The meeting was sponsored by the United States Nuclear Regulatory Commission (USNRC) in collaboration with the Committee on the Safety of Nuclear Installation (CSNI) of the OECD Nuclear Energy Agency (NEA) and The Pennsylvania State University.

The objective of the meeting was to bring together experts involved in the different activities related to boron dilution transients, to promote discussion among these experts, and to focus on the technical issues of concern in resolving the safety significance of such events.

PROCEEDINGS OF THE
CSNI SPECIALISTS MEETING ON BORON DILUTION REACTIVITY TRANSIENTS
October 18-20, 1995

CONTENTS

Session IV: Numerical Analysis of Mixing
Thursday, October 19, 1995
Chair: S. Langenbuch

Session V: Mixing Experiments
Friday, October 20, 1995
Chair: D. Ebert

ACKNOWLEDGMENTS

Individuals responsible for the review of submitted papers for this Specialists Meeting included A. Baratta, M. Champ, D. Ebert, J. Hyvarinen, P. Kloeg, S. Langenbuch, J. Reig, and S. Sandervag. A subset responsible for guiding the technical discussions and developing the Meeting Summary included A. Baratta, M. Champ, D. Ebert, J. Hyvarinen, and S. Langenbuch. Mr. J. Reig of CSNI was responsible for the planning of the meeting which was actively supported by the OECD/CSNI/PWG2. Many of the administrative aspects of the meeting organization were handled by Ms. Chris Wilson of the Pennsylvania State University (PSU) Department of Nuclear Engineering. A. Baratta, of PSU, served as the General and Technical Program Chair. Financial support for this meeting was provided by the U.S. Nuclear Regulatory Commission.

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) and the Republic of Korea (12th December 1996). The Commission of the European Communities takes part in the work of the OECD (Article 13 of the OECD Convention).

NUCLEAR ENERGY AGENCY

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

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

This is achieved by:

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

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

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.

MEETING SUMMARY

1. INTRODUCTION

The CSNI Specialist Meeting on Boron Dilution Reactivity Transients was held at State College, Pennsylvania, U.S.A., from October 18th to 20th, 1995. It was hosted by the Penn State University in collaboration with the US Nuclear Regulatory Commission and the TRAC Users Group. More than 70 experts from 12 OECD countries, as well as experts from Russia and other non-OECD countries attended the meeting. Thirty papers were presented in five technical sessions.

The purpose of the meeting was to bring together experts involved in the different activities related to boron dilution transients. The experts came from all involved parties, including research organizations, regulatory authorities, vendors and utilities. Information was openly shared and discussed on the experimental results, plant and systems analysis, numerical analysis of mixing and probability and consequences of these transients. Regulatory background and licensing implications were also included to provide the proper frame work for the technical discussion. Each of these areas corresponded to a separate session.

The meeting focused on the thermal-hydraulic aspects because of the current interest in that subject and the significant amount of new technical information being generated.

2. GENERAL BACKGROUND

The issue of reactivity-initiated accidents is not a new one. Examination of safety analysis reports from earlier reactor designs shows that such concerns were factored in to account from the inception. Recent interest has arisen concerning a particular class of such accidents -- the boron dilution event. Significant programs in France, Sweden, Germany and elsewhere have concentrated on the differing aspects of such events. In Sweden and France, the effort was directed towards understanding the transport and mixing of a diluted slug as it moved through the primary system. In the US, the emphasis was on the probability and consequences of such an event, while in Finland, effort was directed toward understanding of scenarios which could lead to the formation of dilute slugs.

The proposal for this meeting was initiated by the CSNI Principal Working Group 2 (PWG2). The meeting was organized in cooperation with the US Nuclear Regulatory Commission (US NRC), The Pennsylvania State University and the TRAC Users' Group. There was a total of 29 papers plus two open forum discussions. A total of 71 participants from 16 countries participated in the three-day meeting.

The organization of the meeting emphasized response to the following questions:

1. Do boron dilution transients (DBA or beyond DBA) pose a significant safety risk and why?

2. What are the important phenomena and how should they be addressed?

3. Do we know all the processes that can produce or prevent fuel damage or pressure boundary failure and how should these be addressed?

4. What significance does this transient have with respect to high burn-up fuel?

5. Are the events physically reasonable and can they occur? What will be the consequences?

6. What are the uncertainties in the analyses and the experiments?

7. What are the open issues, if any, and why?

8. How should the open issues be addressed by experiments, analysis, plant modifications, procedural changes, fuel improvements or other?

To enhance the communications and dialogue during the meeting, a series of breaks was included throughout the program during which attendees were encouraged to discuss how they would respond to the above eight questions. At the end of the paper presentations, the audience and speakers were divided into four discussion groups. During the breakout sessions, each group was asked to formulate, under the direction of the session chair, responses to the questions. The answers were then presented to the whole group during session seven, Summary and Conclusions.

Each paper was peer reviewed using accepted journal standards by the Organizing and Steering Committee. The Committee consisted of (in chronological order within the program) A. Baratta, O. Sandervag, D. Ebert, M. Champ, J. Hyvarinen, S. Langenbuch, and B. Eendebak. Their efforts were essential to the successful organization and execution of the meeting as well as the development of the meeting conclusions and recommendations to the CSNI. The members of the Committee met after the meeting to summarize the findings and to discuss the general conclusions and recommendations to be put before the CSNI for consideration. Because of the excellent cooperation, these conclusions were quickly developed and put forth to the PWG2 of the CSNI.

The summary of the meeting is arranged in three parts. The first part is the general introduction and technical remarks. The second part is arranged by session and contains a short discussion of the general technical issues as well as a summary of each session written by the session chair. In the final section, the conclusions and recommendations of the Committee are presented.

3. TECHNICAL REMARKS

The overall goal of the meeting was to determine what the current issues were concerning boron dilution initiated reactivity transients. To achieve this goal requires an understanding of the risk imposed by such an event including both its probability and consequence. To gain this understanding, one needs to understand the types of transients that can lead to a dilution event and the formation of a diluted slug. Next an understanding of the processes that cause the slug to move from where it is formed into the reactor core region is required. One needs to be able to quantify the degree of mixing which occurs as the slug moves from the region where it was formed into the vessel and then subsequently into the reactor core. Finally the impact of the diluted slug on the core's reactivity must be understood and quantified. A critical aspect of understanding the process and its likelihood is a quantification of the uncertainty. Next, it is necessary to identify those areas in which additional research is required because of a lack of knowledge or the existence of a large unacceptable uncertainty.

The process of understanding a boron dilution transient may then be broken into (a) identifying those scenarios which can cause a dilution event, (b) understanding the transport and mixing phenomenon that cause the slug to move to the core region, (c) determining the ability to analyze the impact of the slug on the core reactivity, and (d) assessing the impact on the fuel of this change in reactivity and subsequent excursion.

4. SESSION SUMMARIES

SESSION I - ISSUES AND CONCERNS

As the lead off session for the meeting, this session provided the background for the concern over boron dilution events. A total of six papers from five countries reviewed the history of the concerns over boron dilution events These discussions were based on the perspectives of the different countries involved as well as those of regulatory authorities, a utility and a reactor vendor. In general these concerns centered on dilution events during two different evolutions: inadvertent dilution in connection with maintenance or repair, and dilution in connection with recovery after an incident.

To assess the research needed to quantify the risk and consequences of dilution events during these evolutions requires an understanding of the ability to predict the following:

- Formation and size of an unborated slug
- Migration of the diluted slug to the core
- Migration through the core of the diluted slug
- Response of the core both thermal and mechanical.

From the operational standpoint, several of the speakers noted that actions were already underway to reduce the probability of such events and the consequences. There were sentiments expressed, however, that not everything that could be done had already been done and there was a need for further development and research. Because of these concerns and questions, the speakers concluded that there was in fact a continuing need to research the topic of boron dilution. There was almost unanimous agreement that the issues are well defined and known centering on the ability to accurately predict the events discussed above.

SESSION II - PROBABILITY AND CONSEQUENCES

This session examined the risk to a nuclear reactor of a boron dilution event. A total of five papers from three countries presented differing views and conclusions on the probability and consequences of such an event.

Probabilistic Safety Analysis

Utilities and/or regulatory bodies of the three countries represented in this session have performed Probabilistic Safety Analysis (PSA) for internal, and some for external, events. The first results of those studies released between 1980 and 1993 pointed out that dilution initiating events may be a large contributor to global core melt frequency. From these assessments, the main following boron dilution sequences were pointed out. Concerning the external boron dilution, the first main sequence occurs during normal deboration actions if the main off-site power supply is lost.

Following this initiating event, dilution may go on and, if the natural circulation flow rate is low, there is a potential risk for diluted water slug formation in the reactor piping system.

A second sequence of potential reactivity transient may occur if cold water is injected into a loop seal through the reactor coolant pump (RCP) seal. If the natural circulation is sufficiently low, this may lead to a deborated and cold slug formation.

For inherent boron dilution, the main sequences pointed out were: slug formation during reflux condenser cooling mode cooling following a small break LOCA event (for US and Finland), and secondary to primary leakage (for Finland). Studies are in progress in France on these issues.

Consequences

The consequential core damage probability if the deborated slug previously formed is sent through the core is highly dependent on many parameters or phenomena. During the session, several of them were discussed, such as:

- mixing in the slug formation areas,

- size of the slug,
- reactivity rate insertion,
- mixing in hot leg. downcomer, lower head and core,
- neutronic feedback effect,
- initial shutdown margin,
- EOP's required actions and operator behavior.

According to the assumptions taken into account, it has been shown that conclusions may vary in a wide range.

Backfitting

So far, modifications implemented on the plants to cope with these sequences vary from country to country and from plant to plant. In the USA, it is considered that Emergency Operating Procedure (EOP)'s modification in order to avoid slug formation or to ensure its mixing prior to RCP's pump start up are sufficient to fit with the current regulations. In Finland, modification to the EOP's and more accurate instrumentation have been implemented for a proper avoidance of diluted slug formation. In France, EOP improvement, mechanical modifications on applicable systems and the addition of new automatic safeguards have been implemented to reduce the probability of diluted water slug formation to a very low values for the sequences of concern.

Future Studies

A general agreement came out of this session for the need to better understand or assess the following aspects: mixing phenomena, and coupling of thermohydraulic and neutronic calculations. A complementary screening of potential initiating events for diluted slug formation during maintenance operations or during accidental situations may be necessary.

SESSION III - PLANT AND SYSTEMS ANALYSIS

Session III, Plant and System Analysis, was the longest session of the meeting, featuring nine papers of basically two different categories: actual plant analyses, and tools required for such analyses.

Plant analyses of varying extent were presented in five papers, three of which addressed VVER 440 (thermal hydraulics and core response), one covering a large German PWR (core response) and one the ABB-CE system 80 + advanced PWR concept (design certification). The papers discussed both external and inherent dilution mechanisms. Two key items summarize the general findings: (1) for currently operating reactors, boron dilution possesses significant reactivity insertion potential, and (2) reduction of the maximum boron concentration is a technically feasible means of substantially reducing the risk of an unwanted reactivity transient due to diluted coolant.

Four papers dealt with theoretical prerequisites of plant and systems analysis. One of them concentrated on the theoretical background of core neutronics response to dilute slug insertion, two discussed the state-of-the-art of coupled thermal-hydraulic/reactor dynamics code systems (one developed in Finland and another under development in the United States), and one presented some old U.S. work relating to boron transport in BWRs. Capability to perform truly coupled thermal-hydraulic/reactor dynamic (3D) analyses was perceived as a necessity for detailed study of boron dilution scenarios. It was also recognized that the development of such a capability is a very challenging task.

The breakout session discussions resulted in the following conclusions. Boron dilution can pose a significant safety risk, as best evidenced by the large coolant insertion (in the order of 10 to 20% ΔK/K, depending on core temperature) and possible reactivity insertion rates (comparable or larger than rod ejection). This explains the large efforts by many European countries in this field. There is some desire in the US to name the problem "an issue" rather than "a risk." As to mechanisms that can lead to inhomogeneities in primary boron concentration, there is some hope now that most of the key physical mechanisms have been identified: injection from outside the primary, and distillation-like separation in the primary. It was noted that the boiling-condensing mechanisms may play a role also during sump recirculation phase in containments from which decay heat is removed by externally cooling the steel dome and condensing steam in the inside.

Two important phenomena that are not yet fully understood were identified. These are (1) the resumption of natural circulation (which affects the diluted slug mixing while in transit and the core entry velocity), and (2) the response of high burnup fuel to "milder" reactivity transients. Recent RIA testing of high burnup fuel has yielded catastrophic failures at energy depositions in the order of a tenth of the current RIA licensing criterion. Concerning inherent dilution, boron partitioning between boiling water and steam was also discussed, but the alleviation so available was found to be practically insignificant.

Some scenarios analyzed can be considered idealized, but this can also be of advantage, because reliable conservative analyses are warranted where the consequences of a phenomenon can be very severe even if the probability is low. From the regulatory standpoint, the issue of acceptance criteria regarding inherent dilution scenarios is still open. This is because inherent dilution is not an initiating event, but a physical process, which occurs as a consequence of the loss of one of the engineered barriers (primary integrity) against fission product release. This is the reason why the inherent dilution is almost intractable in conventional level 1 probabilistic studies. Traditional RIA criteria that allow for localized fuel failures are clearly unacceptable in such a situation, especially in the light of the current questions on high-burnup fuel RIA performance.

The remaining open technical issues should be addressed by a balanced combination of experimental and analytical work. A proven scaling basis needs to exist for the interpretation of experiments.

In summary, comprehensive plant and system analysis of boron dilution phenomena and scenarios is a very challenging task, but with the groundwork now mapped, there are no difficulties in principle in tackling the issue. Fortunately, a relatively straightforward means of reducing or even eliminating the risks associated with boron dilution also exists. By increasing the shim rod (fixed burnable poison) worth it is technically feasible to reduce the reliance on dissolved boron, perhaps to such an extent that post-scram dilution recriticality potential would be completely eliminated. This approach has not yet been fully implemented, but an advanced PWR design does exist where substantial steps towards this direction have been taken.

SESSION IV - NUMERICAL ANALYSIS OF MIXING

In the session "Numerical Analysis of Mixing" results of CFD codes like COMMIX, FLUENT and PHOENICS were presented to analyze the flow distribution and mixing effects in the reactor vessel. In addition, results of an engineering approach to describe mixing effects were presented.

The COMMIX code was applied to model the whole primary circuit of a 4-loop PWR plant. Two scenarios were analyzed: an RCP start-up in cold conditions transporting a plug from the loop to the core region. This is an isothermal problem. For the second case, it is assumed that in the hot stand-by condition the pump suction line is filled with cold unborated water which is transported to the core by starting the pump.

In the isothermal problem large flow recirculations were found in the downcomer and reduced recirculations were found in the lower plenum. The mixing effects were strongly improved by these flow recirculations. In the hot zero power calculation, the boron mixing was not as good as in the isothermal case. The flow of the cold plug was predominantly downward in the downcomer due to the buoyancy effect. The results appear reasonable, but up to now no validation exists for the model.

The FLUENT code was applied to analyze boron dilution in cold conditions with 220 ppm initial boron concentration. Various cases differing in the amount of unborated water were analyzed, e.g., continuous supply or a volume of $5m^3$, $10m^3$, or $15m^3$, respectively The effect of nodalization was studied using grids with about 10,000 or 80,000 nodes. After mesh refinement more localized plug transport was observed. Complementary calculations were performed using TRAC-PF1. Because the TRAC model of the vessel uses a very coarse nodalization scheme, the boron distribution is more broadened compared to FLUENT results. The comparison of results of various analyses using coarse and fine meshing and varying discretization schemes, shows the important effect of numerical diffusion which artificially improves mixing effects. Nevertheless, the overall results seem to be consistent.

The PHOENICS code is applied for analyzing the flow distribution in the vessel of a VVER-440 plant. Several cases of RCP start-up were analyzed and some preliminary investigations for the initiation of natural circulation conditions. The results from the calculation show that even a small density difference of 0.5% is enough to enforce the mixing of the unborated plug in the downcomer. For validation of the code some tests from the PTS mixing experiments performed at IVO have been analyzed. This analysis indicates that higher order discretization schemes for the convection terms are needed to reduce the effect of numerical diffusion.

All CFD results still show great uncertainties due to sensitivity to the number of nodes and the numerical discretization schemes which determine the numerical diffusion. Additional important influences on the results are determined by the turbulence models and the specific choice of boundary conditions.

To obtain reliable results, it is necessary to compare CFD results with experimental data from mixing experiments. It was proposed to define a benchmark problem based on experimental data to determine the capability of these CFD codes to study the degree of mixing and the local distribution of the unborated water plug. On this basis, limit values of mixing could be determined to evaluate the reactivity effect during boron dilution accidents.

In addition to the CFD approach, results from an engineering approach were presented to analyze mixing behavior during a small break LOCA in PWR. In this analysis the flow path of the plug is separated into sections for which the dominant mixing effects are determined and for which the degree of mixing is estimated by engineering correlations. It was discussed that both approaches are complementary to evaluate the complex mixing behavior.

The opinion was expressed that the CFD codes will be efficient tools to analyze mixing conditions in the reactor vessel, based on validation by scaled mixing experiments

SESSION V - MIXING EXPERIMENTS

Five papers presented in Session V dealt with experimental measurements or calculations in support of experiments on the boron mixing phenomena. Two companion papers, one experimental and the other an analysis of certain aspects of boron mixing in a scaled thermal-hydraulic test facility, were presented by University of Maryland (UMCP) researchers. The UMCP, B&W scaled test facility was modified to study the generation, transport, and mixing of diluted water under several simulated transient conditions. Early results, from tests conducted at the time, were presented. Results showed that boron-dilute water volumes can be produced in the OTSG and cold leg loop seals of the facility as a result of boiling-condensing mode operation after a simulated SB-LOCA . An analysis study used the CFD codes FLUENT and COMMIX-1C to investigate the importance of geometric scaling differences in the downcomer region between the test facility and the plant.

A different type of boron mixing experiment was performed at the Pennsylvania State University (PSU) using their real-time neutron radiography facilities for visualization of fluid flow through various geometrical configurations. Results indicated that complicated flow patterns can be seen in real-time applications and could be documented for future use in comparisons with computer analysis. Both the work at UMCP and PSU are being supported by the USNRC.

A series of experiments have been and are continuing to be performed at the BORA-BORA vessel mock-up facility in France. These experiments have been run to gain insight into the boron mixing phenomena for a number of potential transient scenarios as well as to provide data for the validation of the N3S computer code for this phenomena.

A scale model of a three-loop Westinghouse PWR has been built at Vattenfall Utveckling AB and used to experimentally investigate rapid boron dilution transients. Research work has been mainly experimental, but recently computational work, using the PHOENICS computer code, to simulate boron mixing in a full scale reactor has been performed.

In a break-away session following the presentation of the papers, the eight questions of interest were discussed and a brief summary of the conclusions is presented here.

1. Do boron dilution transients (DBA or beyond DBA) pose a significant safety risk and why?

These transients may pose a significant safety risk, but there is not enough knowledge at this time to assess the situations. However, precursor events have already occurred in several countries. There are large uncertainties in the knowledge base of the three major phases of the transient progression: dilution generation, slug transport and mixing, and reactivity/power response.

The consensus of the session group was that the studies should continue, and in a more coordinated effort in order to reduce the uncertainties in the knowledge of the several phenomena. These studies have and will continue to produce spin-off effects into other areas of interest, such as new facility data for code assessments of computational fluid dynamics (CFD) and systems codes.

2. What are the important phenomena and how should they be addressed?

As indicated in the response to the first question, there are three major phenomena: dilution generation, slug transport and mixing, and reactivity/power response. These phenomena should be addressed through a combination of experiments and analyses. It was felt that understanding the mixing phenomena was the most important phenomena and should be investigated through experiments and analysis using CFD codes. It was felt that if it is demonstrated that mixing is "sufficient" then the study of the other phenomena could be limited.

3. Do we know all the processes that can produce or prevent fuel damage or pressure boundary failure and how should these be addressed?

4. What significance does this transient have with respect to high burn-up fuel?

These two questions were not discussed due to insufficient time available.

5. Are the events physically reasonable and can they occur? What will be the consequences?

It was agreed that the events were physically reasonable, could occur, but the consequences of these events were highly uncertain. It was noted that there exists a wide variety of reactor designs throughout the world, so that no generic statements could be made and the consequences would be plant or type specific.

6. What are the uncertainties in the analyses and the experiments?

There are many experimental as well as analytical uncertainties. Examples of experimental uncertainties range from disturbance of the flow by measuring probes, limited number of measuring devices, sampling time limitations, fidelity of modelling flow regions such as in the lower plenum, scaling issues, etc. As noted above, there is a wide variety of plant designs and it is difficult to make generic statements. There is a need to perform both integral and separate effects experiments.

7. What are the open issues, if any, and why?

The main open issue is the need to demonstrate that the possible existence of a large amount of clean water does not pose a safety risk to the reactor and to demonstrate that there is a sufficient amount of mixing. However, once again, this issue would have to be addressed on a plant/type specific basis.

8. How should the open issues be addressed by experiments, analysis, plant modifications, procedural changes, fuel improvements or other?

It may be possible, and has been implemented in a number of plants, to implement procedures/hardware fixes to minimize the probability of a boron dilution transient. However, these may be in conflict with other operational procedures with different conflicting goals. It may be possible to design away the boron dilution issue through the increased use of burnable poisons, but this may have adverse impacts on other design goals.

The open issues are being addressed presently, to a certain extent, by experiments being performed in several countries. It would seem to be extremely difficult to perform experiments coupling reactivity/power response with the space and time dependent boron distribution as it moved through the core.

5. CONCLUSIONS AND RECOMMENDATIONS

The general conclusions from the meeting as well as the recommendations to CSNI/PWG-2 together with the specific summary and conclusions from each session are included in the following points.

1. Boron dilution initiated reactivity accidents are a potential safety risk. However, there exists a large uncertainty as to the magnitude of the risk. The potential for this type of accident is not new and has been recognized for many years.

2. It is believed that all of the mechanisms associated with boron dilution have been identified.

3. The most significant phenomena associated with boron dilution are the degree of mixing which occurs as the diluted slug is transported from its source to and through the core, the size and origin of the diluted slug, and the core response to the slug as it moves through the fuel.

4. To assess the risk associated with boron dilution events, the sources of diluted coolant should be investigated, and how the plant behaves during various postulated scenarios such as following pump restart or reestablishment of natural circulation following a boiler-condensation mode during a small-break LOCA scenario.

5. It is important to understand the phenomenology associated with a boron dilution event, such as mixing, when assessing the consequences of some BWR events such as ATWS.

6. Because various specific design features control the degree of mixing and the core response during a dilution event, it is not possible to make a general generic assessment of procedures, design, scenarios, and operating conditions. However, it may be feasible to make such a generic assessment for each type of plant. Depending on the results of these assessments, plant-specific analysis may also be needed.

7. Possible ways to minimize the problem of boron dilution were discussed such as the elimination of soluble boron by having a large shutdown margin through the use of fixed burnable poison or more control rods or by increasing the negative fuel coefficients. These potential design changes may have drawbacks as well, and further effort is needed to assess these methods.

8. In some countries, the industry has already initiated research efforts which have resulted in procedure and plant modifications.

9. High burnup fuel does not add any new phenomenon to the consideration of boron dilution initiated reactivity accidents. However, the degree of burnup and the fuel response should be considered with respect to the potential for core damage.

10. A large uncertainty exists in the quantification of mixing, quantity of diluted water, and core response. In computational fluid dynamics the uncertainty involves numerical diffusion, noding, and turbulence modeling. In experiments, it is associated with the scaling, modeling (e.g., lower plenum) and measurement fidelity.

11. Similar uncertainties are also evident in the systems codes. Here there is a need for increased familiarity with how to perform these analyses as well as to undertake sensitivity studies.

12. Experiments are useful to not only assess the codes, but to increase our understanding of the basic phenomenon.

Recommendations to CSNI/PWG2

From the Specialist Meeting and its conclusions, the Organizing and Steering Committee recognizes that: One of the primary uncertainties is an understanding of how these events can occur. Of equal concern, is a quantification of the uncertainty in the analysis of these events, particularly the mixing and core response.

Therefore, the group proposes to PWG2:

1. Continued efforts should be directed towards the identification of scenarios which could result in boron dilution events particularly during maintenance evolutions. The latter point is important in light of the French experience with accumulators.

2. A series of benchmark exercises for both Computational Fluid Dynamics and Integral System codes should be developed to aid in understanding the problems and uncertainty associated with analysis of potential boron dilution events.

3. Wide distribution of experimental results is needed for assessment of code modeling capability. These experimental results should be used as a basis for International Standard Problems (ISPs) addressing pumped and natural circulation mixing. Such ISPs would best be experiments. The objective of these problems would be to quantify the uncertainty, to investigate the effect of numerical methods, particularly numerical diffusion, and to determine the effect of turbulence modeling on the results.

Boron Dilution Reactivity Transients
A Regulatory Perspective

R. C. Jones, USNRC

Introduction

The purpose of this paper is to summarize NRC actions and concerns relative to boron dilution reactivity transients. These transients are characterized by events which may lead to the development of a unborated (or low borated) slug of water within the reactor coolant system (RCS) which is subsequentially transported into the reactor vessel. The passage of the unborated slug through the reactor core may lead to a reactivity transient sufficient to cause extensive core damage.

The NRC efforts have focused previously on two reactivity transients. These are

- A boron dilution during plant startup
- A boron dilution following a small break loss-of-coolant accident (SBLOCA)

These are addressed separately below.

Startup event

This event sequence starts with the highly borated RCS being deborated as part of a startup procedure. The reactor is in hot shutdown condition with the reactor coolant pumps (RCP) running and the shutdown banks removed. If a loss of offsite power occurs, the RCPs and the charging pumps will trip and the shutdown banks will scram.

When the diesel generators startup, the charging pumps are restarted and charging continues until the volume control tank empties. Because of the very low decay heat during startup, it is postulated that there is little natural circulation in the RCS and the water injected by the charging pumps will undergo little mixing and will accumulate in the cold leg piping and the lower plenum of the reactor vessel. Upon recovery of power, the RCPs are restarted resulting in the transport of the unborated water through the reactor vessel. This scenario was addressed in NRC Information Notice (IN) 91-54, "Foreign Experience Regarding Boron Dilution."

The NRC has studied this scenario as part of its ongoing studies of risk during shutdown operations. Reference 1 provides a detailed discussion of the analyses performed for this scenario; Reference 2 discusses the importance of this event as part of the NRC's study of shutdown operations. In summary, these studies concluded that the probability of such a startup transient is on the order of 10^{-5} per reactor year. However, the consequences are highly dependent on the mixing behavior of the unborated slug. If minimal mixing occurs, significant fuel damage may occur. Our examination of the expected mixing behavior indicates that the lower plenum concentration is only 400-600 ppm less than that originally in the core. Such a decrease in boron concentration will not result in a reactivity excursion in the core.

SBLOCA Scenario

SBLOCA events cover a wide range of plant behavior with larger size breaks providing sufficient energy removal to depressurize the RCS to small break sizes wherein energy removal through the steam generators is required to ultimately depressurize the RCS. It is during these smaller size breaks that steam, produced by the core, is condensed in the steam generators. Due to the low volitity of the boron, the steam condensed will be essentially unborated. The unborated water which collects in the steam generators would ultimately be transported to the reactor core upon recovery of natural circulation, or use of the RCPs to aid in the recovery of natural circulation.

The staff has evaluated this sequence in detail during our licensing interactions on the ASEA Brown-Boveri/Combustion Engineering (ABB/CE) System 80+ design. References 3 and 4 discuss the evaluations performed by ABB/CE on this scenario and the staff's review findings, respectively.

Our review of these studies indicate that recovery of natural circulation is unlikely to lead to core damage from the reactivity transient. However, starting or "bumping" of the RCPs could lead to a large reactivity transient. Whether core damage occurs is highly dependent on the mixing which will occur in the reactor vessel. As noted in Reference 3, substantial mixing will occur and it appears unlikely that a significant reactivity transient would occur for the ABB/CE System 80+ design.

With this information, we reviewed the emergency operating guidelines for all the U.S. pressurized water reactor (PWR) designs. From a review of the ABB/CE and Westinghouse guidelines, it was concluded that the restart of the RCPs is not specified to recover natural circulation. Hence, we believe that a large reactivity transient is unlikely. The Babcock and Wilcox (B&W) guidelines use RCP "bumps" to recover natural circulation. B&W is investigating the potential consequences of reactivity initiated event using these guidelines.

Conclusion

The NRC has investigated the more important scenarios and has concluded that these events do not pose an immediate safety concern. This conclusion was based on our review of the probability of the event, review of the emergency procedure guidelines, and an assessment of the expected mixing.

Notwithstanding these conclusions, several uncertainties exist which warrant further evaluation. These include:

- Effect of recent concerns on the failure behavior of high burnup fuel (References 5 & 6). In evaluating the consequences of any reactivity excursion, the apparent lowered fuel failure threshold for higher burnup fuel rods needs to be considered.

- Most studies performed have been done piecemeal, i.e., the thermal-hydraulics and neutronics evaluations are done separately. This has lead to the postulation of conservative conditions and provides a pessimistic assessment of the consequences of such a reactivity insertion event. Coupled 3-D thermal-hydraulic and neutronic calculations are needed to better estimate the event consequences.

- Finally, calculated mixing of the unborated water within the RCS, in particular, the reactor vessel, appears to indicate that significant reactivity excursions are unlikely. Substantial uncertainty exists in these calculations, and ongoing experimental testing programs should provide valuable insights on our understanding of these events.

References

1. U.S. Nuclear Regulatory Commission, "Probability and Consequences of Rapid Boron Dilution in a PWR: A Scoping Study," NUREG/CR-5819, June 1992.

2. U.S. Nuclear Regulatory Commission, "Shutdown and Low Power Operation at Commercial Nuclear Power in the United States," NUREG-1449, September 1993.

3. U. S. Nuclear Regulatory Commission, Certification of System 80+, NUREG-1462, Vol. 2, August 1994.

4. Technical Paper, "Boron Dilution Reactivity Transients, An Overview of Efforts," Specialist Meeting on Boron Dilution Reactivity Transients, State College, PA, U.S.A., October 18th-20th, 1995.

5. NRC Information Notice 94-64: Reactivity Insertion Transient and Accident Limits for High Burnup Fuels, August 31, 1994.

6. NRC Information Notice 94-64 Supplement: Reactivity Insertion Transient and Accident Limits for High Burnup Fuels, April 6, 1995.

BORON DILUTION REACTIVITY TRANSIENTS: RESEARCH PROGRAMMES AND REMAINING ISSUES

M. REOCREUX
Institut de Protection et de Sûreté Nucléaire
Département de Recherches en Sécurité
CEN- Cadarache, France

ABSTRACT

In recent years reactivity transients due to boron dilution event have become an important matter for reactor safety. Research programmes are being carried out in order to provide the necessary knowledge for predicting these accidents. A review of these programmes and of the issues still remainig is given in this paper. In this review, safety issues for boron dilution reactivity transients are examined first. Reasons for initiating research programmes can then be derived from the actions taken for solving these issues. A list of the phenomena to be described has been drawn up and the corresponding research programmes described. A certain number of questions are raised on unsolved subjects which may serve as a framework for the specialists in this meeting for determining what can be considered as known and what future activities should be performed.

1. INTRODUCTION

The cause considered in this paper for "boron dilution accidents", is an accumulation somewhere in the circuit, of a slug made up of boron diluted water, following various malfunctions. This slug spontaneously or following some operator actions, may migrate and reach the core region where it will induce a reactivity injection. Large core power excursions will result and may lead, at first, to fuel rod failure, and if the power pulse is energetic enough, to fuel ejection. In this latter case, energetic fuel coolant interaction may occur and the consequences of the accident may become, at that time, major. Consequently, since boron dilution is a potential initiating event for large reactivity transients and moreover since the occurrence of the reactivity accident of Chernobyl, boron dilution questions have become an important matter for reactor safety.

Therefore, several studies have been carried out on Nuclear Power Plants (NPPs) in order to define potential scenarios and to determine the safety issues. These studies have shown the needs for several research programmes in order to improve the knowledge and prediction of physical phenomena occurring during such events. As in all on-going research programmes, some questions which have been raised, have not yet become clear and certain technical issues still remain.

This paper will review and discuss these research programmes and the remaining issues. It will try to give a framework to the deeper analysis which will be provided by the papers presented during this 3 day conference. Consequently this paper neither claims to examine all the issues nor to reveal new findings which might solve all problems. Its sole objective is to contribute to the elaboration of a status of what can be considered as known, and what future activities should be carried out in this area in order to improve safety.

2. SAFETY ISSUES AND DEFINITION OF RESEARCH PROGRAMMES

2.1. Safety issues

Several causes are likely, which may initiate a NPP event consisting in the formation of unborated slug somewhere in the circuit, followed by its migration to the core. Once this initiating event is considered, the subsequent consequences are almost entirely determined: the unborated slug while going through the core will originate an increase of reactivity and consequently a power excursion which, depending on the energy release, may induce rod failure, fuel ejection, fuel coolant interaction, mechanical threats on internal structures and on the vessel,.....

Plausible scenarios in which boron dilution problems may lead to reactivity transients are therefore characterized by their initiating event. Analysis of various plant transients are used in order to define the scenarios to be considered but very often one relies on risk analysis studies to discern the most plausible ones. Papers from : EUILLET et al [1995], ATTARD [1995] and DIAMOND et al [1995] give examples of such an approach. For the scenarios obtained, the safety issue is aimed at the potential consequences and it can be summarized by the following question:

Can the plant sustain the energy pulse which may be generated by the unborated slug going through the core?

As it is often proceeded for safety questions, the first approach which is applied to solve this safety issue is a conservative one: one assumes first a "conservative" slug i.e. overestimated in size and with a pessimistic shape. By coupled neutron kinetics and thermal-hydraulic calculations, the amount of reactivity injected by the slug going through the core is evaluated and the resulting power increase derived (see for example [LONGO et al,1995]). The figures obtained will show, very often without making any fuel rod behavior evaluation, that the consequences will be limited or, on the contrary, that they will lead to severe core damage and will be considered as "unacceptable". The results of these conservative approaches are also sometimes presented in the form of the determination of the critical size of the slug above which prompt criticality is obtained. Examples of such evaluations are given in [TRICOT, 1995] and [CLEMENTE et al, 1995].

For the particular event in which only limited consequences are obtained, the conservative approach ensures the validity of the result. For the second case of so called "unacceptable" consequences, there are two possibilities for actions:

-One may go first towards a best estimate approach or at least towards a less conservative evaluation, for example by taking into account the mixing effects between unborated and borated water, or by performing better neutron-kinetics and thermal-hydraulics calculations. This new approach will undoubtedly decrease the consequences of the accident which may become acceptable: in this case the issue is solved. Unfortunately, the consequences may still remain "unacceptable", and additional actions must be taken.

-These additional actions are part of the second category of actions which could be initiated, immediately following the conservative approach and sometimes in parallel to the preceding best estimate approach. These actions consist in finding improvements in the safety systems or in developing improved procedures in order to decrease the probability of the concerned event. Examples of such actions initiated by the EDF in France and which lead to hardware modifications and modifications of operating procedures in French plants are given in [FEUILLET et al, 1995]. The objective is most often here, to reject ultimately this event in the residual risk. This is, as we have seen before, the only way (when the best estimate approach has been used) to cope with consequences which are considered to be too great.

Finally within the general and usual objective of relating research programmes to safety issues, it appears clearly here that the need for using best estimate approaches in order to cope with safety issues as explained above, is the main incentive for proposing and developing research programmes.

2.2. Definition of research programmes

The problems raised by the "boron dilution reactivity transients" are well bounded technical problems in which a systematic and coherent scientific approach of safety questions can be easily followed.

The first step of this approach starts where the real problem lies, that is the plant transients we are asked to evaluate in order to answer the safety issues. Contrary to design basis accidents where the transients are defined in advance, the transients to be studied must be determined by studying the potential initiating events. But every initiating events should not be necessarily examined, only those in which the risk studies have shown that they are of sufficient probability (see [ATTARD ,1995], [FEUILLET et al, 1995] and [DIAMOND et al, 1995]). Of course this classification of transients by risk studies has to be considered with sufficient flexibility, and particularly in order not to miss any important and generic technical problem. In addition to this and still contrary to design basis accidents, operating procedures are established and implemented for many of these transients. These specific procedures may lead to specific physical situations in which the actual behavior of the plant has also to be evaluated.

The second step in defining research programmes consists in the determination of the physical phenomena to be described. This determination is obtained from the analysis of plant transients (examples of such transients can be found in [KANTEE et al, 1995], [ANTILA et al, 1995], [KERESTZURI et al,1995]) particularly transients which have been selected before by risk studies, and from the analysis of the physical situations to which the procedures lead.

The next steps are then quite usual. We must first proceed to physical modelling based on separate effect experiments, if needed. The physical models are introduced in computer codes. These codes must be validated against experimental data, specifically with the view of being able to make transposition to NPPs. Experimental programmes for obtaining the necessary data base for validation should be developed. They should include separate effect tests for the physical phenomena which are unsufficiently known and also integral tests to check the overall capabilities of the code in coupling complex phenomena and in making a transposition to the plant.

3. RESEARCH PROGRAMMES: AN OVERVIEW

3.1. Physical phenomena to be described

The first and most important step in defining research programmes is the determination of the phenomena to be described. Four categories of phenomena can be distinguished : the formation of the unborated slug, the transport of this slug to the core, the transfer of the unborated slug through the core and the resulting reactivity insertion, and finally the fuel response to the reactivity transient.

3.1.1. Formation of the unborated slug

The formation of the unborated slug is the first key phenomena as it is at the initiation of the whole transient by determining initial boundary conditions. Many different events may produce this slug but physically, one can distinguish two different situations:

In the first situation, the fluid is a single phase liquid in the circuit or liquid/air (for example during refilling phases). Accumulation of diluted water will result from the injection into the primary circuit of unborated water, for example from the pump seal or from leaks between secondary and primary circuits. To obtain accumulation in single phase flow, the flow rate must be almost zero which means for example that natural circulation has stopped in the loop concerned. Single phase flows at very low flow rate especially in the range of the transition between natural circulation and stagnant flow are then the phenomena to be described. Transport of boron will determine in which conditions the slug can develop. Some mixing phenomena will occur at the front between borated and diluted water but as the flow rates are very low and almost zero, mixing will be quite moderate.

In the second physical category of situations, the fluid is two-phase in the circuit (steam and liquid). In the same way as in single phase flow, accumulation of diluted water may result from different injections of unborated water into the primary circuit, but, in two phase flow, there is an other possible source of unborated water which is the pure water obtained from steam condensation. This case occurs for example during the reflux condenser mode in PWR where the steam condensed in the steam generator may accumulate in the intermediate leg. In order to obtain accumulation, the flow velocities near the slug should be almost zero. But contrary to single phase situations, this does not necessarily mean that flow rates are very low everywhere in the concerned loop as far as complex two phase flow patterns may occur with countercurrent flows, phase stratifications, complex heat and mass transfers. Differences in boron transport between liquid and steam will privilege liquid phase and can give consequently particular transport modes. Mixing effects will include in addition to the single phase ones, those specific to two-phase flows. Finally the phenomena to be described here are the two phase flows mainly in the range of low flow rates and boron transport including some mixing effects typical of two phase flows.

3.1.2. Migration of the diluted slug to the core

The transport of the diluted slug towards the core can be initiated by various operator actions, for example by restarting a pump or restoring natural circulation. The dynamics of this flow rate reactuation are governed by the usual thermal-hydraulics in single phase or two phase flow. The key phenomenon, here, is the behavior of the slug during its transport by the flow. Diffusion of boron and hydrodynamic mixing will be the phenomena which will modify the boron concentration distribution inside the slug. These phenomena will occur of course in the pipes but will be more accentuated when the slug enters the downcomer and afterwards the lower plenum. This accentuation is due first to the strong changes of geometry: from 1D for the pipe to 2D for the downcomer and finally to 3D for the lower plenum where in addition to this a great number of internal structures are located. It will be also reinforced by the contribution in the downcomer and in the lower plenum of the fluids coming from the other loops which may supply significant additional mixing. The transport of the slug will depend on the nature of the flow, single phase or two phase. The two-phase flows in particular will add specific phenomena such as the "turbulence" effects induced by the moving interfaces between steam and liquid or effects such as heat and mass transfer between fluids having different thermodynamic characteristics.
Finally, mixing phenomena will ultimately determine the shape of the slug entering the core (more or less extended in total volume, with a large or small radius and a small or large height respectively). They will also give the boron distribution inside the slug (sharp or distributed profile).

3.1.3. Migration through the core of the diluted slug

The hydrodynamic of the slug inside the core is governed by phenomena of the same type as those in the downcomer and in the lower plenum, i.e. 3D flows including mixing with the neighbouring fluids. In the core, these phenomena have some specifities such as cross flows between subchannels, effect of mixing grids,....which are similar, for example, to the ones in thermal mixing in DNB processes.

The low values of boron concentration in the slug provoke an injection of reactivity. For this reactivity injection the neutron kinetics phenomena are the usual ones but due to the 3D characteristic of the slug, they will also be 3D. As the slug is moving through the core, the reactivity characteristics will change which means that the 3D thermal-hydraulics will react strongly on the 3D neutron kinetics. Inversely, the power increase due to the reactivity injection may modify the flows which may provoke a feedback from neutron kinetics on thermal-hydraulics. 3D thermal-hydraulics and 3D neutron kinetics phenomena are consequently strongly coupled during this phase.

3.1.4. Thermomechanical response of the fuel rods

The reactivity transient induces a power pulse in the fuel rods. Rod degradation may then occur due to several effects such as fuel temperature increase, internal rod pressure increase, fission product release, rapid clad temperature increase when DNB occurs, rim effect,..... The rod degradation depends directly on the power pulse strength and is therefore usually related to the amount of injected enthalpy. Different steps are generally distinguished for increasing values of enthalpy reached. These are: no rod failure, rod failure without fuel ejection, rod failure with fuel ejection (and eventually FCI occurrence in the subchannels). For each of the transitions between these successive steps, criteria have been set up in the framework of the Design Basis Accidents which give values of level of enthalpies in order to avoid entering into the next degradation step. Conservative criteria to avoid rod failure or fuel ejection have been set up this way. Questions have been raised recently about the validity of these criteria when applied to high burn up fuels. It appears in fact that depending on the burn up, the relative influence of phenomena on the thermomechanical response may change, and that new ones may also appear. The thermomechanical reponse of rods is the last phenomena which should be examined if we wish to be complete in the description of boron dilution reactivity transients. However they are not often considered as part of the boron dilution studies because it is felt that they are covered by other reactivity accidents studies. This is certainly true but these phenomena should not be forgotten in order not to miss any possible specificities.

3.2. Research programmes

1/ Modelling of thermal-hydraulic behavior of NPP single phase or two phase slug formation

As we have seen in the identification of the phenomena, a large part of the evaluation of this phase relies on single phase or two phase thermal-hydraulics. Consequently standard codes can be used provided boron tracking is added. This is achieved by adding a supplementary transport equation. As it is generally considered that there is no practical feedback between the boron concentration and the fluid parameters computed by the thermal-hydraulics code, there are no real theoretical difficulties (only programming difficulties may be encountered). Moreover as the diluted slug generally forms in a pipe, 1D models, at least for the slug itself, are sufficient. Mixing at the boundary of the slug may not be very important and in several cases not very precise modelling is certainly sufficient. The main difficulties for the description of this phase, are located in the flow regimes one has to predict. In single phase flow they are regimes which are situated between natural circulation and stagnant flow. Furthermore in two phase flow they comprise complex two phase regimes with stagnant flow, countercurrent flows, very low velocities,.....For these regimes, present thermal-hydraulic codes are not always well adapted. Research has often to be carried out on numerics (instabilities, excessive computing times). Physical laws also need to be better assessed in this low or zero velocity domain. Experiments must be run in order to investigate the plant response in this area. In fact large programmes on code development, code assessment and on system experiments are devoted to this domain in the general effort on safety thermal-hydraulics. For example the programmes on large systems loops like BETHSY, ROSA, PKL have several tests providing data in single phase or two phase, for natural circulation, effects of thermal stratification, performance with balanced or unbalanced loops, and for simulation of various operator actions. All these programmes contribute effectively to the modelling of this phase and need to be actively continued.

2/ Modelling of single phase thermal-hydraulics of slug transport to the core

When the flow is restored in a loop where a boron diluted slug has formed, one has to describe the hydrodynamics behavior of the slug in 1D, 2D, and finally 3D geometry. For this purpose, 3D single phase codes are available and can be used provided boron tracking is added (see for example the use of the COMMIX code in [SUN et al, 1995]) These codes do not raise generally large problems regarding their basic equations [SUN et al, 1995] . More or less detailed description of 3D geometries are available using, for example, finite element techniques. Nevertheless the application of these codes raises a certain number of problems:

-In order to obtain sufficient accuracy (or just to be converged in the tracking of the boron front) a very large number of nodes is generally required (more than several thousands: see for example [ALAVYOON et al, 1995] and [GANGO, 1995] with PHOENICS code). Consequently computer time may become excessive. Controlling and optimizing computer time efficiency is then an important research topic which, in certain cases, conditions the success itself of the overall studies.

-Another important problem on which research effort has been made and continues is the control of the numerical diffusion. A numerical diffusion too large will shade out the actual physical diffusion and mixing. As the boron concentration front behavior is the key phenomena in the overall mixing problem, solving the numerical diffusion question is essential. In order to answer this question, several numerical techniques are proposed or are being developed in ongoing research actions ([KYRKI-RAJAMAKI, 1995]).

-Physical laws for boron diffusion and mixing are introduced in the codes (see for example [GANGO, 1995]). In single phase flows they raise few theoretical problems (strength or weaknesses of turbulence models are known, and quite extensive experience exists in implementing these models in flow models). However a strong need is generally expressed for validating these laws in representative situations. This validation represents one of the main physical modelling efforts in the research activities related to boron dilution reactivity transients. It will have to rely on specific experiments (see below).

3/ Modelling of two phase flow thermal-hydraulic of slug transport to the core

The problems for modelling the slug transport to the core when the flow regime in the NPP is two phase phase are similar to the ones identified in single phase flow but with largely increased difficulties:

-First, two phase flow codes with 3D capabilities are very few and experience in their use is quite restricted.

-The existing 3D two phase flow codes should be largely improved in their capability for allowing fine topological description.

-The control of numerical diffusion is also a major problem but it is largely more difficult than in single phase because not all numerical methods can withstand the specificities of the two phase flow numerical behavior (instabilities, non linearities,...). Some research programmes dealing with this subject have been initiated (for example [MACIAN et al., 1995] with TRAC PF1).
-Mixing effects in two phase flow, as discussed before, have several causes and some of them are typical of two phase flow ("2 phase turbulence"). On this point a lot has to be done and few studies are devoted to it.
It appears finally that existing two phase codes need to be largely improved if a more accurate evaluation than the one we can perform, is required. Extended experimental data bases will be needed for the development and the assessment of the constitutive laws in the representation of 3D flows with fine meshing first and secondly of the mixing models in two phase. To our knowledge, almost nothing has been done in this area with the exception of some limited 3D two phase flow studies.

4/Experimental research programmes for mixing models validation

If we summarize a boron dilution reactivity transient we find:
-At one end the slug formation the size and content of which, is mainly governed by external events: therefore the uncertainties on physical models in this phase are certainly not the predominant ones.
-At the other end there is the flow through of the slug in the core for which the calculation of the consequences on reactivity does not highlight significant physical uncertainties.
-In between there is the transport of the slug to the core where the mixing phenomena are, physically speaking, the most uncertain and the ones which may influence the consequences of the accident the most.
This explains why an important experimental effort is being made in order to obtain data for assessing these mixing models.

Several experiments are being conducted ([GREEN et al ,1995a], [GREEN et al , 1995b], [ALVAREZ et al, 1995], [ALAVYOON et al, 1995]) which represent the plant geometry and in which mixing effects are simulated. These system experiments require a scaling down process, which is most often not easy and which requires specific care. A complete representativity of the real case is often not possible and compromises have to be made (see for example the discussion by ALVAREZ et al [1995]). The consequences of such constraint on the scaling capabilities of the models have to be carefully analysed and checking the models in different experiments in which different compromises have been chosen, should certainly be recommended. However this very large research effort covers almost only single phase flow situations. If two phase flow transients were to be considered, similar experiments would have to be initiated with all the additional difficulties that two phase flow studies generally produce ,of which we are all aware.

5/Development of the coupling of 3D neutron kinetics codes and 3D thermal-hydraulics codes

In order to obtain a representative best estimate evaluation of the reactivity transient when the diluted slug is going through the core, 3D codes are necessary for neutron kinetics and thermal-hydraulics. These codes exist and have been developed and assessed in other frameworks. It seems that no specific developments of the individual codes are needed for boron dilution questions except perhaps some special care concerning numerical diffusion for thermal-hydraulics, but this is covered by research programmes we have already discussed earlier. The problems arise from the need to couple these codes (see for example the coupling of RELAP5 with PANBOX2 by JACKSON et al [1995] and the modified version of TRAC PF1 used by IVANOV et al [1995]).

The coupling problems are mainly numerical problems (exchange of variables, time steps control, convergence,...). They are also computing time problems as these codes are per se, very large codes, and their coupling sometimes adds with increasing factors computing times which, individually may already be prohibitive. Some research in solving the numerical problems is being done as is absolutely required for running any practical calculation. For optimising the calculation, to our knowledge very little has been done or has just started. Research efforts on this point should probably increase in the future with the use of new computer architectures and particularly those using parallel processing.

6/ Development of fuel rod thermomechanics codes

In order to be complete, fuel rod thermomechanics responses to reactivity transients induced by the boron dilution events should be evaluated. In fact, very often only the criteria discussed precedently on rod failure and fuel ejection are used. If more details were required, one would necessarily have to go to the fuel rod thermomechanics codes. These codes are in fact developed within the framework of the rod ejection design basis accident and no specific research is being done on boron dilution questions. One can only note here that substantial research has restarted in this area on code development as well as on experimental support, especially in order to investigate the high burn up effects. One can also note that these codes should be coupled with thermal-hydraulic codes due to various interactions such as DNB occurrence, fission gases release especially with high burn up fuel, interaction with molten fuel,.....

4. REMAINING ISSUES

It is certainly premature at the beginning of a specialists' meeting to draw up a significative list of the remaining issues, since this should be precisely one of the main conclusions of the meeting. Nevertheless, before discussions during this 3 day meeting will have shed light on the issues where questions still remain open, we can already define, in the interrogative form, the areas which raise problems and which are certainly part of the remaining issues to be discussed.

One of the main areas where significant problems remain concerns the thermal-hydraulics. When boron transport has been added to the usual single or two phase codes, one has potentially all what one needs for carrying out accident evaluation. The questions which still remain concern in fact the capabilities of these codes:
-Are the codes sufficiently validated in the low velocity range when the regimes are between natural circulation and stagnant flow?
-Is the validation which has been made in other safety areas sufficient for boron dilution reactivity transients?
-Are the numerical capabilities of the codes in these regimes sufficient in terms of robustness, accuracy and computing time efficiency?
-Are the present codes sufficient regarding the topology they cover, 1D, 3D, and especially regarding the fine noding required for tracking boron fronts?
All these questions are valid for both single and two phase codes but we must notice that they are undoubtedly more accentuated for two phase codes.

The second area where issues remain is the modelling and testing of mixing phenomena. This, as we have seen, is a key area for boron dilution transients. In addition to some of the thermal-hydraulics issues raised before and which also apply to mixing, there are specific physical and numerical questions which can be summarized in the following way:

-Are the present codes sufficient in terms of diffusion and turbulence modelling? We must observe that this question is certainly more pertinent for two phase flow than for single phase flow.
-Are these models sufficiently validated and what are their exact scaling capabilities?
-Are the existing or planned experiments sufficient for validating the mixing effects in the codes? Do we need to explore two phase situations?
-Is numerical diffusion well controlled and do we have methods which ensure in all cases and without significant drawbacks, an efficient numerical diffusion control?
-Do the present numerical methods allow reasonable computing times for the accuracy we would like to have?

The last area concerns the core response on neutronics and thermomechanical aspects.Two main issues probably remain in this area:
-In the coupling of 3D neutron kinetics codes and 3D thermal-hydraulic codes are there urgent needs in improving computing time efficiency as computer capabilities are increasing very fast and as calculation cost is continuously decreasing?
-The developments which are made in other frameworks regarding 3D thermal-hydraulics and fuel rod thermomechanical behavior during reactivity transient, are they sufficient for boron dilution accidents? Do we need additional code development and additional experiments for their assessment to cover exactly the range of parameters encountered in "boron dilution accidents"?

The list of research issues which has been drafted here, has been widely open as we explained before and is only a prospective one. Presentations and discussions during this meeting will certainly help in specifying issues and hopefully will give orientations for future activities in order to resolve them.

5. CONCLUSIONS

The research programmes carried out for investigating boron dilution reactivity transients are relatively well defined programmes regarding their objectives and their relations to NPP safety problems. Starting from the scenarios to be studied on the plant, physical phenomena have been identified for which research programmes have been defined including physical modelling, code development, code validation and performance of experimental programmes. A review of these research programmes has been made which shows that they are covering the main phases of the transient. Most critical phenomena are investigated: thermal-hydraulics code validation in the range of low velocity flows for the slug formation, development of 3D thermal-hydraulics with boron tracking for the slug transport, performance of mixing experiments for validating these codes, coupling of 3D neutron kinetics and 3D thermal-hydraulics codes for describing slug flow through the core and the consequent reactivity insertion, fuel rod thermomechanical code development and assessment. Obviously unsolved issues still remain in these research programmes. A list of questions related to these issues has been established and discussed. From this list, one can see that most of them can be answered by pursuing research activities. But it appears also that research issues are not the only issues in the general safety problem. Events related to boron dilution reactivity transients, using even the most advanced and best estimate evaluation might nevertheless produce consequences that would be difficult to cope with. For these events research is no longer the first issue. The only way to obtain a solution will be to improve the plant systems or operator procedures. For these actions, research findings are undoubtedly of great help. But it is clear that these improvements are here the key issues. Finally, it appears that handling the boron dilution reactivity transients requires certainly simultaneous efforts, first in the direction of plant improvements and management and second in the direction of an improvment of the general knowledge of phenomena by pursuing adequate research activities.

REFERENCES:

F. ALAVYOON, B. HEMSTROM, N.G. ANDERSSON, R.I. KARLSSON, Epenmental and Computational Apprach to investigating Rapid Boron Transient Boron Dilution Tmasients in PWRs; CSNI Specialist Meeting Session V, Penn State, Oct. 1995

D. ALVAREZ, F. CASES, A. MARTIN, S. STELETTA, Boron Mixing Transients in a PWR Vessel. Physicl Studies; CSNI Specialist Meeting Session V, Penn State, Oct. 1995

M. ANTILA, P. SILTANEN, R. KYRKI-RAJAMAKI, T. VANTOLA, Analysis of Core Response to the Injection of Diluted Slugs for the Loviisa VVER6440 Reactor; CSNI Specialist Meeting Session III, Penn State, Oct. 1995

T. ATTARD, Boron Dilution Reactivity Transients. A Regulatory Perspective; CSNI Specialist Meeting Session II, Penn State, Oct. 1995

M. CLEMENTE, S. LANGENBUCH, Determination of Reactivity Insertin Effects due to Local Boron Dilution; CSNI Specialist Meeting Session III, Penn State, Oct. 1995

D.J. DIAMOND, P. KOHUT, H. NOURBAKHSH, P. SECKER, K. VALTONEN, Probability and Consequences of a Rapid Boron Dilution Sequence in a PWR; CSNI Specialist Meeting Session II, Penn State, Oct. 1995

J. FEUILLET, J.J. GOETTER, Boron Mixing Transients in a PWR. Safety Analysis. CSNI Specialist Meeting Session II, Penn State, Oct. 1995

P. GANGO, Application of Numerical Modeling for Studying Boron Mixing in Loviisa NPP; CSNI Specialist Meeting Session IV, Penn State, Oct. 1995

J. GREEN, J. FLOYD, K. ALMENAS, M.di MARZO, Multi-dmensional Modeling of Boron Dilution Transients in the UCMP 2X4 Facility; CSNI Specialist Meeting Session V, Penn State, Oct. 1995

J. GREEN, K. ALMENAS, M.di MARZO, Generation, Transport and Mixing of Boron-Dilute Waterin a Scaled Integral Model of a B&W Reactor; CSNI Specialist Meeting Session V, Penn State, Oct. 1995

K. IVANOV, R. MACIAN, A. IRANI, A.BARATTA, Analysis of Reactivity Tmasients using a Coupled 3D Thermal-hydraulic Kinetics Code; CSNI Specialist Meeting Session III, Penn State, Oct. 1995

C. JACKSON, H. FINNEMANN, Explanation of Core Physics Phenomena during a Boron Dilution Transient; CSNI Specialist Meeting Session III Penn State, Oct. 1995

H. KANTEE, S. SAVOLAINEN, Scoping Study of Inherent Boron Dilution for Loviisa Nuclear Power Plant; CSNI Specialist Meeting Session III, Penn State, Oct. 1995

A.KERESZTURI, A. GACS, M. TEIBISZ, Calculation of Start up of an Inactive Coolant Loop with Lower Boron Concentration; CSNI Specialist Meeting Session III, Penn State, Oct. 1995

R. KYRKI-RAJAMAKI, M. RAJAMAKI, Boron Dilution Transient Calculation with the HEXTRAN code using CFDPLIM; CSNI Specialist Meeting Session III, Penn State, Oct. 1995

J.LONGO, F.L. CARPENTINO, C.B. MARTIN, A Consevative Approach on Small Break LOCA Boron Dilution; CSNI Specialist Meeting Session III, Penn State, Oct. 1995

R. MACIAN, J.H. MAHAFFY, Implementation of a High Order Boron Tracking Algorithm with Turbulent Diffusion in TRAC-PF1. Application to the Study of Boron Dilution Trabsients in PWR; CSNI Specialist Meeting Session III, Penn State, Oct. 1995

J.G. SUN, W.T. SHA, Analysis of Boron Dilution in a Four Loop PWR; CSNI Specialist Meeting Session IV, Penn State, Oct. 1995

N.TRICOT, Boron Dilution Events. First Investigations and Safety View;.CSNI Specialist Meeting Session II, Penn State, Oct. 1995

Local Boron Dilution - Swedish Regulatory Concerns

Oddbjörn Sandervåg
Swedish Nuclear Power Inspectorate (SKI)

Presented at the CSNI Specialist Meeting on Boron Dilution Reactivity Transients

PSU Oct 18-20, 1995

Contents of the presentation

Historical Background

Main Safety Issues

Safety Assessment

Research in Sweden

Preventive Measures

Remaining Issues

Historical Background

In safety analyses of boron dilution transients for PWRs it is normally that the boron is uniformly distributed within the reactor coolant system

In connection with startup of one of the PWRs in Sweden the potential for accumulation of zones with lower boron concentration in the loops was identified by the utility

Scoping studies were conducted by the utility, and it was found that there could be a potential for large reactivity insertions

Although the probability was considered small, administrative preventive measures were taken in the plant

The Swedish concerns were reported internationally in late eighties.

The Chernobyl accident triggered review of reactivity insertion accidents also for other types of reactors

Main Safety Issues

The main safety issue is the potential for a large reactivity insertion that could cause excessive fuel damage, prevent safe cooling of the core or overpressurize the primary system.

Two main categories:

Inadvertent dilution in connection with maintenance or repair

Dilution in connection with recovery after an incident

The Swedish efforts have been concentrated on the first item

Safety Assessment

The first approach when dealing with a potential safety problem is to identify the obvious, easy-to-defend, limiting conditions and phenomena as a basis for a conservative assessment.

This is in principle possible for local boron dilution, for example

- A stagnation zone in a loop is needed

- If the amount of water is small enough, nothing serious will happen

However, scoping studies have shown that a very small amount of clean water may lead to significant reactivity insertions

This creates at least two problems:

The sufficient amount of clean water is so small that a large number of scenarios are possible as initiators, and it is difficult to establish confidence that all have been identified

Mitigation of the consequences of these scenarios would significantly affect other safety functions of the plant

The strategy from the Swedish side, utility as well as regulatory, is to identify a larger minimum clean volume so that the a design basis for changes can be established and the problem can be dealt with

Our analyses indicate that this strategy can be successful, and that the prevention measures are adequate

Research in Sweden

The research objectives are to investigate the function of the protective barriers against occurrence of RIA

Theoretical studies have been carried out and also underway study mixing of diluted and undiluted volumes as they are transported to the core.

An experimental program is also carried out as support of the theoretical efforts

Theoretical studies are also conducted to explore reactivity consequences of clean water reaching the core. Models are being implemented and tested

Results so far indicate that the uncertainties in the assessment may still be significant

Remaining Issues

The uncertainty is high

Establishment of adequate assessment tools, validation of models

Control of numerical diffusion, and simulation of turbulent mixing

Scaling of the phenomena observed

Confidence has to be established in the methodology

Application of the methodology to full size reactors

Scenarios of Boron Dilution Incidents and Concerns Regarding Plant Safety

M.Simon, GRS mbH, Germany

1 INTRODUCTION

The results of PSA in France, in the United States and in the Scandinavian countries, concerning non-power operation of PWR, initiated investigations with the aim to study the applicability of these results for German PWR. The reference plant for the investigations which have been performed within the framework of the accident management (AM) program[1] was Biblis, Unit B, a plant which has been also the reference plant for the German risk study. The study has been performed by GRS in co-operation with the utility /1/, /2/, /3/ and mainly focusses on two areas: loss of RHR events and boron dilution events during low power or shutdown (LPS).

2 IDENTIFICATION OF PLANT OPERATIONAL STATES (POS)

Compared with the normal plant state, which is called 'power operation mode', the plant goes through different modes of operation during a refueling outage. These different modes of operation can be classified as POS. They have been described to some extent in the plant operation manual of the reference plant. Each type of POS is defined in terms of plant conditions, such as coolant temperature and pressure, inventory level and boron acid concentration. A specific POS ends if one of these plant conditions changes considerably with time. For this study, in all 13 POS were identified for a normal refueling outage (shutdowns for maintenance purposes have not been considered). Table 1 gives a general view on these POS. As the negative reactivity value of the control rods is not

[1] Sponsored by the Federal Minister for the Environment, Nature Conservation and Nuclear Safety (BMU)

sufficient for cold shutdown of a PWR the boron concentration must be considerably increased for shutdown and reduced again during plant start-up.

Table 1 : Identification of Plant Operational States (POS) During LPS

POS	Description	Demanded Boron Acid Content (ppm)
POS 1	Reactor shutdown (scram)	C = cr
POS 2	Cooldown with SG to 275° C	C = cr + 400
POS 3	Cooldown with SG to 200° C	C = cr + 800
POS 4	Cooldown with SG to 110° C	C = 2200
POS 5	Cooldown with RHRS to 50° C	C = 2200
POS 6	Draining RCS to mid-loop	C = 2200
POS 7	Fill of reactor cavity for refueling	C = 2200
POS 8	Refueling	C = 2200
POS 9	Draining reactor cavity to mid-loop	C = 2200
POS 10	Refill of RCS and pressurization to 32 bar	C = 2200
POS 11	RCS heatup with RCP to 120° C	C = 2200
POS 12	RCS heatup with RCP to 260° C	C = 2200
POS 13	Reactor startup to criticality	C = cr

C = boron acid concentration (ppm)
Cr = actual boron acid concentration

3 SYSTEM CONFIGURATIONS DURING DIFFERENT POS

The availability of systems for core cooling and boron addition to the coolant significantly changes with different POS. This is because safety systems have to be blocked to prevent inadvertent actuations, e.g. the high pressure emergency core cooling system (ECCS) and the accumulators, or because certain systems are no longer physically effective, e.g. the steam generators (SG), or because there are reduced requirements for the number of available trains, e.g. the residual heat removal system (RHRS). As a consequence, the number of available systems can be significantly reduced during certain POS, especially during POS 5 to POS 10. Furtheron, these modes require instrumentation and supervision which are not necessary during power operation, e.g. the mid-loop instrumentation and the control against deboration of a subcritical reactor.

Regarding the boron acid content in the reactor cooling system (RCS), the normal systems providing chemical blending of the coolant are switched off during certain POS (POS 6 to POS 9), and the content of storage tanks (e.g. RWST and accumulators) is used for filling processes. Failure of instrumentation, wrong boron acid concentration in systems or tanks, or faulty operator action can cause dilution of water in the RCS with the possibility of unintentional reactor criticality.

Table 2 shows the availability of systems for boron addition to the coolant during different POS for the reference plant.

Table 2 : Plant Operational States (POS) and System Configuration (available systems for boration of RCS)

	Shutdown					Refueling				Startup			
System	POS 1	POS 2	POS 3	POS 4	POS 5	POS 6	POS 7	POS 8	POS 9	POS 10	POS 11	POS 12	POS 13
	C= cr+ 400	C= cr+ 800	> 2200	C= 2200	C= 2200	C= 2200	C= 2200	C= 2200	C= 2200	C= 2200	C= 2200	C= 2200	C= cr
HP ECCS	4	4	4								4	4	4
ACCU	4	4	4	4						4	4	4	4
RWST							4	4					
CVCS	2	2	2	2	2	2			2	2	2	2	2

C = Boron acid concentration (ppm)
HPECCS = High Pressure Emergency Core Cooling System
RWST = Refueling Water Storage Tank
ACCU = Accumulators
CVCS = Chemical and Volume Control System

4 INITIATING EVENTS DURING LPS CONDITIONS

To identify initiating events (IE) during LPS conditions, operating experience (OE), existing studies and theoretical considerations have been used. By this approach it should be ensured that all important incidents that have occured and other possible scenarios are considered in the study. The sources of OE were the German licensee event reports and the incident reports of the IRS of OECD / IAEA. The sources for possible scenarios were the French PSA for PWR 900 MW and PWR 1300 MW and the German risk study phase B.

The evaluation of international OE on the basis of the IRS indicates a relatively high frequency of events with loss of decay heat removal and also with boron dilution compared to the rest of IE during LPS. As a result two main groups of IE were identified:

- IE which affected the cooling of the fuel
- IE which caused deviations from the required boron acid concentration of the coolant (dilution).

Additional IE have been identified, such as cold overpressurisation of the RCS or loss of inventory from the RCS, but they are not discussed in this presentation.

The identified IE for unintentional dilution during LPS are shown in table 3.

Table 3 : Initiating Events for Unintentional Dilution During LPS

IE	Initiating Events
B1	**Maintenance work with demineralized water**
B2	**Addition of low borated water from RWST or accumulators**
B3	**Uncontrolled boron dilution by feedwater from unplugged SG tubes**
B4	**Uncontrolled boron dilution from the chemical and volume control system**
B5	**Improper boron dilution during preparation for core criticality**

One of the characteristics of these IE is that they are strongly related to the different POS, which means that a specific IE can only occur during specific POS. Another important characteristic is that the physical conditions, the available time for operator actions and the availability of systems are different in case that a specific IE would occur during different POS. A matrix was constructed which shows the relationship between IE and POS for loss of RHR events and also for dilution events. The matrix was filled with IE gained from OE. Engineering judgement was used to find out if IE could also occur during other POS, independent from OE (table 4).

Table 4 : Matrix of Initiating Events (IE) For Dilution

	Shutdown					Refueling				Startup			
IE	POS 1	POS 2	POS 3	POS 4	POS 5	POS 6	POS 7	POS 8	POS 9	POS 10	POS 11	POS 12	POS 13
	T ≤ 290 °C	T ≤ 275 °C	T ≤ 200 °C	T ≤ 110 °C	T ≤ 50 °C	drain RCS to mid-loop	fill for refueling	refueling	drain RCS to mid-loop	T ≤ 90 °C	T ≤ 120 °C	T ≤ 260 °C	T ≥ 260 °C
B1						OE			OE				
B2						X	OE		OE	X			
B3									OE				
B4	X	X	X	X	X					OE	OE		OE
B5													OE

OE = operating experience, X = possible events

B1 = Maintenance work with demineralized water
B2 = Addition of low borated water from RWST or accumulators
B3 = Uncontrolled boron dilution by feedwater from unplugged SG tubes
B4 = Uncontrolled boron dilution from the chemical and volume control system
B5 = improper boron dilution during preparation for core criticality

5 SELECTION OF SPECIFIC INITIATING EVENTS FOR BORON DILUTION

IE with dilution of the coolant inventory can be subdivided in two different groups:

- events with slow deboration of the coolant,
- events with rapid deboration of the coolant.

5.1 Slow Dilution

From the OE only IE with slow dilution are known. IE with fast dilution have been hypothetical up to now. For slow and homogenious dilution large amounts of demineralized water are needed for unintentional core criticality. The POS where such large amounts of water are added to the RCS are:

- shutdown of the plant from hot stand-by to cold shutdown,
- filling the reactor cavity for refueling,
- startup of the plant from cold shutdown to criticality.

The investigation has shown that as a result of wrong maintenance work at systems connected with the RCS only small amounts of demineralized water may intrude into the RCS. These amounts of water are not large enough to substantially reduce core subcriticality. Therefore, regarding slow and homogenious dilution, the following IE have been selected as initiators for unintentional core criticality:

- Uncontrolled dilution from the chemical and volume control system during shutdown (B4/POS 1),
- Addition of water of wrong boron acid concentration from the RWST (B2/POS 7),
- Uncontrolled dilution from the chemical and volume control system during RCS refill (B4/POS 10).

Unintentional criticality during shutdown may result in a fast power transient because there is no further shutdown system available and the power increase is only limited by the Doppler coefficient. The power increase may challenge the safety valves on the primary and/or secondary side as long as the RCS is closed or may cause steaming of the coolant when the RCS is open. Such a reactivity incident happened in a German plant during start-up when the boron concentration was unintentionally reduced to a value lower than calculated for core criticality. The reactor power shot up from zero power to about 10% of nominal power and the SG safety valves opened before the reactor was scrammed by the protection system.

5.2 Fast dilution

Formation of a water plug of pure water in a RCS loop can occur if the RCPs are not running and

- boron dilution continues during plant startup after LOP

- feedwater is spilling into the RCS via a defective SG tube during preparation for plant start-up
- feedwater enters the RCS during a SG tube rupture accident after isolation of the defective SG
- the RCS is cooled in the reflux condenser mode during a severe accident. Condensed steam will collect in the cold loops.

Regarding fast dilution the following scenario has been selected as an initiator for such an event:

- Formation of a water plug in one loop and startup of the related reactor coolant pump (B5/POS 13).

Depending on the amount and boron concentration of the water plug, starting a RCP in such a situation can cause a reactivity accident. Calculations by GRS for such a scenario have shown a strong impact on the RCS because of the fast power generation in the fuel elements resulting in possible fuel element damage and steep pressure increase in the RCS. The results of these calculations will be presented in section III of this meeting.

Table 5 : Initiating Events for Unintentional Dilution During LPS

IE	POS	Description	Speed of event	Recovery actions
B2	7	Filling reactor cavity with water of wrong boron concentration from RWST	slow	yes
B4	1	Injection of demineralized water through the chemical and volume control system during shutdown	slow	yes
B4	10	Injection of demineralized water through the chemical and volume control system during startup	slow	yes
B5	13	Startup of a RCP after improper boron dilution	10 sec	no

5.3 Frequency of IE

The frequency of these selected initiating events was calculated for the reference plant to assess the reliability of hardware, operator actions, and the quality of administrative measures which prevent such IE. For this purpose a fault tree was constructed for each IE and a probabilistic analysis was performed. The results have shown that the calculated frequencies are in the same range as the expected frequencies for rare accidents during power operation.

Other IE, or the same IE but during other PCS, have been assessed to have either less consequences on plant safety and/or to have lower frequencies. Therefore the risk from other IE has been considered as less significant.

6 Lessons learned and conclusions

The evaluation of events during LPS has shown that human performance is a big contributor for these events. However, the evaluation also shows that many events have been supported by poor design or insufficient outage planning. Here are some examples for dilution during LPS:

- unavailabilty of the boron concentration detector caused unintentional dilution of the RCS by the CVCS
- intrusion of feedwater from an unplugged SG tube to the related RCS loop caused dilution of the coolant
- poor ergonomics of the counters (adjustors) for relation between borated water and demineralized water caused wrong blending of injected water by the CVCS. The boron contcentration was unintentionally reduced from initially 2450 ppm to 1500 ppm during plant start-up .

The study has identified possibilities to further reduce the probability of occurrence of events with severe consequences during LPS. For prevention of dilution incidents the following proposals were made:

- reliable instrumentation and alarms for control of subcriticality
- automatic and complete isolation of the chemical and volume control system after loss of forced flow by the RCP (to prevent demineralized water injection into the RCS),
- improvement of the instrumentation for measuring the boron content in the RCS (number of positions, delay time)

- administrative control to prevent start of a RCP after local dilution is assumed or has been identified.

The selected IE have been also investigated regarding possible recovery actions, such as accident management measures. The investigations have shown, that the success of such actions strongly depends on

- the available time,
- the availability of backup systems, and
- the physical conditions during the event.

The operating experience has shown that a loss of core cooling can easily be detected by physical observations (e.g. by RHR pump failure, temperature increase of the coolant) and mitigation measures can be initiated therefore well in time. Also homogenious dilution of the coolant may be detected in time as there is a big margin from a subcritical core to criticality. But a local dilution of the RCS cannot be detected in the existing design, and therefore mitigation measures may come too late or may have no effect. This is especially valid for rapid dilution of the coolant in the RCS (see table 5).

7 Existing and future requirements regarding boron dilution incidents

7.1 Requirements for existing plants

The existing rules and regulations for PWR in Germany focus on the power operating mode, not on the shutdown mode. Regarding reactivity related accidents, mainly accidents by malfunction of control rods have been considered. Regarding boron dilution there is only a very limited number of accidents which has been considered in the design or is included in the procedures of the plant manual :

- faulty dilution via the CVCS during power operation
- possible dilution after SGTR

7.2 Requirements for future plants

For the EPR, which is the common project between France and Germany, the GPR/RSK has recommended in the "Proposals for a common safety approach for future PWR" with respect to boron dilution incidents:

"Thorough assessment has to be done as regards the possibilities of boron dilution accidents in particular during shutdown states. Reactivity accidents resulting from fast introduction of cold or deborated water must be prevented by design provisions so that they can be practically excluded. Among these design provisions, automatic features to avoid inadvertent diluted water slug formation, leak detection devices, supervision of the boron concentration of systems have to be considered to the appropriate extent."

No design accidents and no final technical solutions have been defined up to now to fulfill this requirement, as the EPR project is still in the conceptual phase, not yet in the basic design phase.

8 REFERENCES

/1 / M. Simon:
Evaluation of Low Power and Shutdown Events in German PWRs
CSNI Workshop on Special Issues of Level-1 PSA,Cologne,May 27-29,1991

/2/ M. Simon:
Identification of Adverse Initiating Events in PWR During Low Power and Shutdown States and the Assessment of their Consequences on Plant Safety
IAEA Technical Committee Meeting on Procedures for Probabilistic Safety Assessment for Shutdown and Other Low Power Operating Modes,Arnhem, Nov. 1-4,1994

/3/ M.Simon
Spezielle Risiken und Unfallszenarien im Anlagenstillstand
SVA-Vertiefungskurs : Sicherheit von Kernkraftwerken im Stillstand
Brugg-Windisch, Switzerland, March 29-31,1995

OECD/CSNI Specialist Meeting on
Boron Dilution Reactivity Transients
State College, Pennsylvania, USA, 18-20 October 1995

PERSPECTIVES ON BORON DILUTION

Harri Tuomisto
IVO International Ltd, Vantaa, Finland

I.1 Introduction

The topic of this paper is to discuss perspectives of local, inhomogeneous boron dilution events in the PWR's. When using term "Local Boron Dilution" we mean all events that could lead to formation of partially diluted or completely unborated slugs in the primary system.

Experience and views of this contribution are based on the work carried out at IVO, a Finnish utility owning and operating the Loviisa VVER-440 plant. However, our intent is not to present any 'official' utility perspective, but rather discuss various aspects of local boron dilution. First in Section I, a historical perspective and requirements are discussed. A short overview is given on IVO's activities concerning both the Loviisa VVER-440 plant and some new plant concepts.

Section II of this presentation is devoted to efforts to answer the questions distributed to the authors and participants of this meeting by the Programme Committee in the letter of June 27th, 1995 [1].

Throughout the presentation, a clear distinction is made between external and inherent boron dilution events. **An external dilution** occurs when diluted or pure water slug is created by injection from the outside. Such events could be called as well system-related local boron dilution events. **An inherent dilution** mechanism is connected to a number of accident classes, where dilution could take place through an inherent phenomenon during the accident such as boiling-condensing heat transfer mode inside the primary system, or backflow from the secondary system in case of primary-to-secondary leakage accidents. Additionally, a few remarks will be made concerning boron dilution and reactivity effects during severe accidents. A short introduction of these phenomena was presented at the OECD in the autumn of 1992 [2], which also served to start the process towards arranging this Specialist Meeting.

I.2 Historical perspective of external dilution

The boron dilution transients included originally in the scope of FSAR's dealt solely with homogeneous external dilution. Water of low or zero boron concentration being injected to the primary circuit, was assumed to mix quite perfectly with all the primary coolant inventory.

Reassessment of reactivity accidents was quite independently initiated after the Chernobyl accident in various countries for Western PWR plants: USA [3], France [4] and Sweden [5]. These studies comprised of assessment of external dilution events, with an overall approach to address all pertinent cases. Steam generators, chemical and volume control system, dilute accumulator or diluted refueling water storage tank and diluted containment sump are mentioned as potential sources of diluted water. Dilution may occur during power operation, shutdown or accident conditions. The sequence of events may vary significantly in different scenarios, e.g.:

- pure water from the secondary side may flow to the primary circuit due to maintenance errors during shutdown,
- reactor coolant pumps may stop during startup dilution thus initiating slug formation,
- inadvertently diluted accumulators may leak to primary circuit during power operation or during accident conditions, etc.

It is interesting to note that the French study already mentions the typical inherent dilution mechanisms: boiling and condensing heat transfer to produce pure condensate, and backflow from the secondary side during a steam generator tube rupture accident.

A common feature of the external boron dilution studies is a strong role of probabilistic risk assessment. Probabilistic methods have proven to be efficient when evaluating, to what extent boron dilution has to be prevented.

In Finland, reassessment of boron dilution events was accelerated for Loviisa reactors after actions taken by EdF in France became known. Because of the complex geometry of the primary loops and the main gates valves in both the hot and cold legs, there are various extra aspects of slug formation and transport in VVER-440 reactors [6]. Probabilistic assessment of potential consequences indicated that the risk of a serious reactivity accident was too high. Consequently, various actions were decided to be taken, including changes in automation, operating procedures and equipment. After the performed modifications the frequency of a serious reactivity accident due to an external, local boron dilution is much less than 10^{-6} per reactor year [13].

I.3 Inherent dilution

When primary coolant decreases during such accidents as small break LOCAs, anticipated transient without scram (ATWS) and primary-to-secondary leakage accidents, a boiling/condensing heat transfer mode may be established between the core and steam generators. During this mode, pure (practically zero concentration of boric acid) condensate is collected in the loop seal and a nonborated slug is created. This slug can be transported to the core after re-establishing the natural circulation or starting a RCP in the primary circuit.

The mechanism of inherent boron dilution during various accident scenarios was introduced by Finnish regulatory body STUK to be considered for the proposed PWR concepts for the fifth nuclear power unit in Finland [7]. Accordingly vendors (Siemens for Konvoi like PWR-1400 and AEE for VVER-91) carried out the initial studies. In autumn 1993, the utility application for the fifth unit was canceled by the Finnish Parliament, and these studies did not enter the licensing review.

After more detailed problem definitions, extensive inherent dilution research programs were initiated for Loviisa VVER-440 (by IVO) and for VVER-91 concept (IVO in cooperation with Russian Kurchatov Institute and Gidropress). In this Specialist Meeting, a series of papers will be presented on thermal-hydraulic studies for slug formation and transport at Loviisa during various accidental conditions [10], [11] and [12].

II. SPECIFIC QUESTIONS

II.1 Do Boron Dilution Transients Pose a Significant Safety Risk?

When severe accidents were incorporated into a defense-in-depth strategy in Finland, a specific requirement was defined that the fraction of reactivity initiated core damage sequences should be less than 1% of the total core damage frequency [18]. Meeting this criterion is then interpreted to mean that the safety risk is not significant.

To be able to evaluate the risk for safety, both the consequences and the frequency of potential boron dilution scenarios should be known. A correct estimation of consequences require rather realistic calculations, which means plenty of new research work both in thermal-hydraulics and reactor dynamics.

Reactor dynamic calculations have shown that even rather small deeply diluted or pure coolant slugs (even less than 0.5 m^3, see Ref. 6) may pose a high potential to reactivity excursion, if sufficient mixing does not take place before the slug entering the core. Particularly, the cases with startup of a reactor coolant pump can bring the slug with a high velocity to the core and a severe reactivity excursion might result. Consequently, there is a significant potential for safety risk, if sufficient mixing of the slug cannot be demonstrated. The consequences on the reactivity are more severe in case that slug is pushed through the core by starting a reactor coolant pump than in case of slug transportation by natural circulation [13].

In case of inherent dilution, thermal-hydraulic scoping calculations and experiments clearly show that formation of a diluted slug in the cold leg seal can take place under certain accident conditions. Thus, a potential exists for reactivity consequences. However, during small break LOCAs mixing of the slug in the downcomer seems to bring an effective mitigation to boron dilution [12]. Based on the performed studies for Loviisa plant, a highest potential occurs during

ATWS events (control rods not inserted!), and during large primary-to-secondary leakage accidents, if feedwater is not reliably isolated.

Probabilistic treatment of the external dilution cases has been introduced to the existing PSA level 1 environment e.g. in France and in Finland. In both cases it was found that the safety risk is high, and corrective actions were taken. More accurate treatment of the slug transport would not necessarily remove the need for these corrective actions.

No corresponding risk evaluation is available for inherent dilution. A standard PSA level 1 technique is not easily applicable here, since the boron dilution takes place in the sequences which are normally considered as success sequences. The task can be divided to two levels. The first task is to demonstrate that the sequences with potentially high consequences, such as ATWS and primary-to-secondary leakages (without feedwater isolation), have so low frequency values that they can be screened out to the residual risk group. The second task is to show for events that cannot be screened out (such as boiling/condensing modes during small break LOCAs) acceptability of the consequences. From the phenomenological viewpoint this can be a very demanding task, and it will be discussed in more detail below.

To conclude one can say that external boron dilution may pose a significant safety risk if left unattended. Inherent boron dilution will need more studies, before the risk aspect can be reliably addressed.

II.2 What are the important phenomena and how they should be addressed?

External dilution

Important phenomena to be addressed in external dilution studies are:

- slug formation mechanisms including estimation of boric acid concentration, temperature distributions and slug size,

- transport of the slug by forced (startup of a reactor coolant pump) or natural circulation under single phase, subcooled and possibly stratified conditions,

- response of the reactor core to the diluted or/and colder coolant slug when entering to the core as a homogeneous or strongly inhomogeneous mixture.

Slug formation mechanisms require a thorough plant-specific study which covers all possible dilution events. Engineers with a good process knowledge of the specific plant have to be involved.

Slug transportation evaluations can be done by applying 3D computational fluid dynamics codes. This is an ambitious task, and a lot of work is still needed. Fortunately, current computational methods and computer resources make it possible to perform meaningful analyses. The difficulties lie mainly in modeling the turbulent mixing, and in avoiding numerical diffusion. Ref. 16 gives an example of overcoming these difficulties in application of PHOENICS code. However, the most serious problem is a shortage of representative experimental data for validation of the applied method. Two experimental facilities have been built and experiments started to study the slug transport: BORA-BORA at EdF Hydraulic Laboratory in Chatou [14] and a facility at Vattenfall Älvkarleby Laboratory in Sweden [15]. These facilities are 1:5 -scale transparent models of three-loop PWR's. Similar experiments are being planned for VVER-1000 at Gidropress in Russia. After successful validation of numerical models against these experiments, the CFD approach should give sufficiently reliable results.

Successful analysis of the reactivity effects requires application of 3D reactor dynamics models, because of inhomogeneous boric acid or/and temperature distribution in the core inlet and of necessity to calculate the ramp effects.

Inherent dilution

Slug formation analyses during accident conditions require that all thermal hydraulic two-phase flow phenomena are accounted for. A list of essential phenomena will be discussed by Savolainen in [10]. Onset of the slug transportation has to be evaluated separately. Slug transport involves extra aspects in comparison to external dilution such as saturated conditions, high pressure injections to the cold leg, slow natural circulations and importance of density differences. Core response analyses require a full coupling between thermal-hydraulics and reactor dynamics, since core power generation may accelerate slug motion.

Slug formation can be analyzed using thermal hydraulic system codes, since these are capable of treating the pertinent phenomena. Naturally, their application should be assessed against such integral experiments, where condensate formation has been occurred. Difficulties may arise when analyzing ATWS sequences, where coupling of the primary circuit behavior to the core response is very complex.

Example of addressing slug transport during inherent dilution will be presented by Gango [12] using the PHOENICS code. The application to this problem was validated against thermal mixing experiments, which have been made to study thermal mixing of high pressure injection during potential pressurized thermal shock sequences of the Loviisa reactor vessels.

For the reactivity calculations, 3D reactor core dynamics code HEXTRAN dynamically coupled with thermal hydraulic system code SMABRE, will be used in Finland [13].

Important phenomena are those related to slug formation, slug transportation and core response. These phenomena should be addressed with methods capable of realistic modeling.

II.3 Completeness of the knowledge

As already indicated above there are some areas, where the knowledge base is not yet complete. Plant specific studies are required for completion of slug formation both for external and inherent dilution cases. The same applies to studies of onset of slug motion.

Slug transport phenomena are basically known to a degree to create good understanding, but the modeling capability is still widely insufficient. Validation of 3D methods is a basic deficiency. Correct modeling of turbulent mixing is essential, and effects of numerical diffusion have to be avoided. Fully-coupled 3D reactor dynamics and primary system thermal hydraulic code packages exist (e.g. HEXTRAN-SMABRE), and others are under development.

To conclude, the knowledge base is sufficiently complete to start an extensive study of the boron dilution problem. The actual calculation methods still require further development and validation, before plant-specific analyses can be considered reliable.

II.4 Significance with respect to high burnup fuel

Different limits were compared when considering acceptance criteria for boron dilution events at Loviisa. In regard to boron dilution, the reactivity margin from achieving recriticality to the RIA limit 963 J/gU_2O (230 cal/gU_2O) is not very wide [17]. In principle, a similar conclusion should apply to other reactor concepts as well. The given RIA limit is applicable to the external boron dilution events. In case of inherent boron dilution, fuel may have suffered of various loadings due to initiating accident, and lower RIA limits should be applied.

For high burnup fuels the acceptable RIA limits may be much lower. Therefore the results of boron dilution analyses for lower burnup fuels are not readily applicable. Since the reactivity margin from recriticality to a serious reactivity accident is not very wide, it is not, however, expected that utilization of high burnup fuel would be excluded because of boron dilution events. Depending on the applied acceptance criteria, boron dilution events have to be reanalyzed for high burnup fuels.

It is not to be expected that boron dilution would exclude utilization of high burnup fuel. The acceptance criteria have to be, however, reconsidered and additional boron dilution analyses may become necessary.

II.5 Are the events physically reasonable and can they occur? What will be the consequences?

Interestingly enough, part of an inherent dilution mechanism was studied for Loviisa in an extensive experimental program already in years 1985-1990. At that time the main interest was in a potential boron precipitation process. The conclusions were very plant-specific, since they were affected by arrangements of low and high pressure safety injections, reactor and reactor vessel features, utilization of borax in the ice of ice condenser etc. The result of the study states very clearly that it is not physically reasonable to assume boron precipitation in the Loviisa core during any loss-of-coolant accidents [19].

The final answer to the inherent boron dilution problem should be given just in these terms: it is not physically reasonable that inherent boron dilution problem would lead to a reactivity accident with unacceptably high radioactive releases.

The risk evaluation can be effectively divided to system related probabilistic studies and to the estimations of phenomena related expectancies. The risk is acceptable if it can be shown that either the sequence can be positively excluded by screening based on a very low frequency, or it is not physically reasonable to assume that the considered scenario would lead to a serious reactivity accident. In some cases the latter may turn out to be a complicated task. Analogous situations in severe accident field are efficiently resolved by applying the ROAAM (Risk-Oriented Accident Analysis Methodology) [20], which is also applied for Loviisa studies [18]. In complex situations of inherent boron dilution, it may turn out necessary to apply this methodology, since it can give a structured and traceable answer to the question of physical reasonability.

No doubt the phenomena leading to boron dilution are physically reasonable. More profound studies, closely coupled to the risk evaluation, are needed to answer the question, whether it is physically reasonable to assume that a boron dilution event can lead to large radioactive releases.

II.6 Uncertainties in analyses and experiments

Uncertainties are involved in many levels of boron dilution studies. When analyzing the risk aspects, system-related probabilities are bound very often to human actions, since many sequences take place during shutdown conditions. Such uncertainties are to be treated with the level 1 PSA methods.

Uncertainties in physical phenomena are related to the issues already discussed above. Development of numerical modeling and results from slug transport experiments can surely decrease considerably the associated uncertainty level. For the complex treatment of the uncertainties in the overall studies, the ROAAM method can be applied.

II.7 Open issues

Open issues related to physical modeling of boron dilution events were already discussed above.

Including the ATWS sequences to the scope of inherent boron dilution is one of the open questions. Potential for severe disturbances in the core power is very high during ATWS conditions, since the reactor core may still be critical when inherent boron dilution occurs. However, the frequency of ATWS is very low and the probability screening should be applicable.

Inherent boron dilution may take place during severe accident sequences, if secondary side bleed&feed becomes available after onset of the core heatup. Scoping studies for Loviisa confirm this result [11]. Thus, boron dilution may have implications to severe accident management strategies. Typically, secondary bleed&feed is a preferred means to arrest the severe accident progression. It has to be evaluated, whether boron dilution would have an impact on this preference.

The acceptance criteria for the consequences of different boron dilution events have not yet been defined.

Other boron dilution related issues of severe accident management are mentioned in Appendix 1. If water is injected to the primary circuit or to the containment in order to terminate the core melt progression or to cool the molten core, subcriticality of the degraded and corium should be evaluated.

As mentioned in Appendix 1, boron concentration related issues may also arise in BWR's, if boron injection is applied as an ATWS management measure. Treatment of BWR's has been, however, left outside the scope of discussion here.

II.8 How should the open issues be addressed by experiments, analyses, plant modifications, procedural changes, fuel improvements or other?

Based on the above discussions, some recommendations concerning the open issues can be readily given.

Slug formation from external sources should be prevented by all practical means. That includes plant modifications (boric acid concentrations in dilution tanks, automation to prevent inadvertent injections etc.) and procedural changes.

Slug formation during inherent processes should be mitigated by procedural means, and if necessary with applicable plant modifications.

Slug transportation is bound with uncertainties due to insufficient modelling capability. Therefore, efforts should be continued to obtain experimental data base on forced and natural circulation transport phenomena, and the analytical 3D CFD tools should be developed and assessed to give realistic and reliable calculational results.

Changing the reactivity control in such a way that boron would not be utilized any more as a neutron poison solved in primary coolant, would naturally be an ultimate measure against boron dilution events. Considering the existing knowledge base and developments to carry out reliable analyses, as well as efficacy of performed plant modifications, it does not seem necessary to make such a radical change to existing fuel management practices.

In regard to severe accident management aspects of diluted boron concentrations, further research is necessary to estimate, what are the remaining problems and if these have implications to applied severe accident management strategies.

REFERENCES

[1] OECD/NEA letter EN/S/1354 of June 27th, 1995 for the authors of CSNI Specialist Meeting on Boron Dilution Reactivity Transients, by Mr. J. Rieg.

[2] H. Tuomisto, Reactivity considerations of severe accidents, Presented at SESAM Meeting on October 14 and at PWG2 Meeting on October 15, 1992.

[3] D.J. Diamond, C.J. Hsu, R. Fitzpatrick, Reactivity accidents: A reassessment of the Design-Basis Events, Brookhaven National Laboratory, NUREG/CR-5368 (BNL-NUREG-52198), January 1990.

[4] A. Gouffon, Developments concerning reactivity accidents in PWRs, OECD/CNRA Mtg. on Regulatory Requirements and Experience Related to Low Power and Shutdown Activities, 1991. With appendices:
J.M. Lanore, M. Champ, Probabilistic safety assessment of French pressurized water reactors: Importance of non-power states, IAEA-SM-321/45, and
M. Champ, Risk evaluation of dilution accidents, Report S.A.C.P. No 122, Dec. 1990.

[5] S. Jacobson; Risk evaluation of local dilution transients in a pressurized water reactor, Linköping University, Studies in Science and Technology, Dissertations No. 275, 1992.

[6] P. Siltanen, M. Antila, Extended protection against reactivity accidents caused by slugs of diluted water in the Loviisa reactors, 2nd Symposium of AER, Paks, Hungary, 21-26 September 1992.

[7] J. Hyvärinen, Thermal-hydraulic safety aspects of horizontal steam generators. Proceedings of International Seminar on Horizontal Steam Generator Modeling, March 11-14, 1991, Lappeenranta, Finland, Vol. II, pp. 327-340.

[8] K. Haule, J. Hyvärinen, Potential reactivity problems caused by boron separation/dilution during partial loss of inventory transients and accidents in PWR's, STUK/YTO Memo (12 November 1991).

[9] J. Hyvärinen, An inherent boron dilution mechanism in pressurized water reactors, Nuclear Engineering and Design **145** (1993) 227-240.

[10] S. Savolainen, Overview to boron dilution problem in Loviisa NPP, OECD/CSNI Specialist Meeting on Boron Dilution Transients, State, College, Penn., 18-20,1995.

[11] H. Kantee, S. Savolainen, Scoping study of inherent boron dilution for Loviisa Nuclear Power Plant, ibid.

[12] P. Gango, Application of numerical modeling for studying boron mixing in Loviisa NPP, ibid.

[13] P. Siltanen, M. Antila, R. Kyrki-Rajamäki, T. Vanttola, Analysis of core response to the injection of diluted slugs for the Loviisa VVER-440 reactors, ibid.

[14] D. Alvarez et al., Three dimensional calculations and experimental investigations of the primary coolant flow in a 990 MW PWR vessel, NURETH-5, Salt Lake City, Sept. 1992, Vol. II, pp. 586-592.

[15] H. Tinoco et al., Physical modeling of a rapid boron dilution transient, Vattenfall Utveckling Report VU-S 93:B21, 17.9.1993.

[16] P. Gango, Numerical boron mixing studies for Loviisa Nuclear Power Plant, NURETH-7, Saratoga Springs, NY, USA, Sept. 1995.

[17] P. Siltanen, Boron dilution management: Proposal for acceptance criteria, IVO Report, 23 September 1993 (in Finnish).

[18] H. Tuomisto, In pursuit of consistency and completeness in the severe accident assessment and management, OECD Specialist Meeting on Severe Accident Management Implementation, Niantic, Connecticut, USA, 12-14 June 1995.

[19] J. Tuunanen, H. Tuomisto, P. Raussi, Experimental and analytical studies of boric acid concentrations in a VVER-440 reactor during the long-term cooling period of loss-of-coolant accidents, Nuclear Engineering and Design **148** (1994) 217-231.

[20] T.G. Theofanous, "Dealing with phenomenological uncertainty in risk analysis", Workshop I in Advanced Topics in Reliability and Risk Analysis, Annapolis, MD, October 20-22, 1993.

Appendix 1

REACTIVITY CONSIDERATIONS OF SEVERE ACCIDENTS

1. REACTIVITY INITIATED SEVERE ACCIDENTS

1 a Inadvertent boron dilution (local) during shutdown states of PWR's

1 b Boron dilution in PWR's due to boiler-condenser mode operation during SBLOCAs, transient or SBLOCA initiated severe accident sequences, ATWS accidents etc.

1 c Boron dilution due to inverse flow conditions during SG tube rupture (single or multiple) accidents

1 d Power oscillations of BWR's, e.g. ATWS initiated by power oscillations

1 e Power instability due to inadvertent LPI during BWR ATWS accidents (after initial boron injection or even with no boron control of ATWS)

2. REACTIVITY CONSIDERATIONS DURING SEVERE ACCIDENTS

2 a Restoring the core cooling injection in BWR's (PWR's have to be checked also) after control rod meltaway during severe accidents, leading to neutron power generation

2 b When control rods melt away from the BWR core, molten material drops to the water pool on the lower head - this can create small scale FCI phenomena that can accelerate a non-borated water slug through the reactor core and cause a rapid power excursion.

2 c Subcriticality of the corium
- analyses
- monitoring

WESTINGHOUSE ASSESSMENTS OF REACTIVITY TRANSIENTS DUE TO RAPID BORON DILUTION

Toby Burnett, P.E., Westinghouse Nuclear Energy Systems

Presented at the CSNI Specialist Meeting on Boron Dilution Reactivity Transients State College, PA 18-20 Oct 1995

Introduction

In this discussion, we define a rapid boron dilution event as the sudden addition of a volume of boron free or nearly boron free water into the core region. Rapid boron dilution events can be postulated for which the reactivity addition in the core could be localized and potentially severe.

Westinghouse first analyzed a rapid boron dilution event nearly 25 years ago (ref 1), for a startup of an isolated loop containing cold, clean water in a PWR with loop isolation valves. Point kinetic analyses were found to grossly overpredict the severity of the transient. Transient three-dimensional analyses were completed, and showed about 3% fuel rod failure and less than 0.5% of fuel elements completely melted. Total energy release was determined to be insufficient to breach the Reactor Coolant System (RCS). We considered these results to be unacceptable despite stringent administrative controls on opening loop isolation valves, and added protection-grade interlocks circuits to prevent inadvertent startup of an isolated loop. The event was thus removed from design basis evaluation and further analyses became academic. Therefore, analyses of dilution events in our Final Safety Analysis Reports (FSARs) have been limited to gradual and uniform dilution events, which do not involve local reactivity effects.

Much more recently, in 1992, Westinghouse participated with the Electric Power Research Institute (EPRI) in Outage Risk Assessment and Management (ORAM) studies, during which we studied potential rapid boron dilution transients. Our findings were published in the EPRI report, "Risk of PWR Inadvertent Criticality During Shutdown and Refueling", NSAC-183, December 1992, by Toby Burnett, et. al. The remainder of this paper is extracted from section 3 of that source.

Our judgement stated in that report was very similar to the conclusion we had reached decades earlier: The consequences of starting a reactor coolant pump (RCP) with a large volume of unborated water in its suction cannot be proven to be generically acceptable. Indeed, we stated our belief that cases can probably be hypothesized that can be proven to have unacceptable results. Therefore, for the events we've assessed, we continue to believe that such events must be prevented.

Anatomy of a Rapid Boron Dilution Event

Two things are required for a rapid boron dilution event to occur: the collection of a stagnant volume of relatively unborated water, particularly in the crossover or cold leg of the RCS, and the subsequent rapid transmission of that volume of unborated water into the core region. As discussed in the report "Some Local Dilution Transients in a Pressurized Water Reactor" (ref 7), the formation of a stagnant zone of unborated water requires very low or no flow in the reactor coolant loops.

If any RCP is operating, there will be sufficient circulation throughout the RCS to prevent the formation of a stagnant zone. The Residual Heat Removal (RHR) system provides enough circulation in the portions of the RCS being circulated to prevent the formation of a stagnant zone in the reactor vessel, cold leg piping between the RHR discharge and the reactor vessel, and hot leg piping between the reactor vessel and the RHR suction. However, the RHR system may or may not cause circulation through the steam generators. Any of three conditions may block RHR circulation through the steam generators:

1. Steam generator tubes drained. During most of a refueling outage, the steam generator tubes are drained (filled with air or nitrogen).

2. Vapor pocket. If the RCS is totally depressurized and the water level near the flange, a low temperature water vapor pocket is likely to exist at the top of the steam generator tubes (the manometer effect, ref 9).

3. Thermal gradients. If water in the steam generator shell is warmer than the active potion of the reactor coolant, RHR flow through the reactor vessel is generally too low to force water through the steam generator tubes as a parallel path. This condition would exist if the RCPs had been shut off prior to completion of the RCS cooldown.

If all forced circulation is shut off, natural circulation through the reactor coolant loops as a result of core decay heat will generally result (unless prevented by one of the above mechanisms). Natural circulation flow through the steam generators is generally much higher than would be forced by RHR flow. However, it's possible for variations in temperatures between the steam generators to prevent natural circulation through one loop, particularly at cold shutdown. We therefore recommend that the reactor coolant in the steam generators and reactor coolant suction piping (the crossover legs) should be considered as stagnant at all times that all RCPs are shut off.

If diluted water is added to a stagnant region, a pocket of less borated water results. The addition of diluted water may occur either deliberately, as in the case of proceeding to a normal reactor startup, or inadvertently.

One French study (ref 8) analyzed the potential impact of a clean water slug inserted into the core. Two parametric cases were analyzed, in one case the boron concentration of the core was assumed to decrease by 400 ppm with no change in core inlet temperature, while the other case involved a reduction in core inlet temperature of approximately 110°F in conjunction with the dilution. For the case of the dilution only, no core damage was predicted; however, for the case of the dilution in conjunction with the cooling, significant core damage was predicted.

To gain insight into local effects involving only a few assemblies, scoping calculations were performed by the Westinghouse nuclear fuels division for a few fuel assemblies typical of current design, without burnable poisons, in cold water, both clean and borated (2000 ppm). The reactivity was calculated with the KENO-PC code, using parameters considered typical of current fuel design. The results are listed in the table below.

No. of Assemblies	U-235 Enrichment	Boron Concentration	k_{eff}
1	5 w/o	0 ppm	0.95
2	4 w/o	2000 ppm	0.76
2	4 w/o	0 ppm	1.05
2	5 w/o	2000 ppm	0.81
2	5 w/o	0 ppm	1.08
3	4 w/o	2000 ppm	0.82
3	4 w/o	0 ppm	1.12
3	5 w/o	2000 ppm	0.87
3	5 w/o	0 ppm	1.15
4	4 w/o	2000 ppm	0.90
4	4 w/o	0 ppm	1.20
4	5 w/o	2000 ppm	0.96
4	5 w/o	0 ppm	1.24

These results are reported to illustrate the potential highly localized effects of subjecting high enrichment assemblies to cold, pure water. It should be noted that core designs rarely place two fresh unrodded assemblies next to one another, and that burnable poisons (to whatever extent is required by the core design) would be added before putting the assembly in the reactor core.

In principle, an optimally shaped and placed volume of clean, cold water on the order of 10 ft^3 is estimated to be capable of producing prompt criticality in a cold core at beginning of cycle. Considering the relatively high velocities at which fluid can be swept into the core (roughly 3 feet/second for a single RCP running on a 4-loop plant, twice that velocity on a 2-loop plant), neither prompt criticality nor high reactivity insertion rates can be ruled out. If the reactivity insertion is rapid enough, severe core damage could result, including dispersal of molten fuel with resultant pressure waves. Fuel dispersal has been conservatively calculated for some postulated rapid boron dilution events, but we know of no realistic calculation which proves such a result is possible for any realistic scenario.

Accurate deterministic analyses of the possible extent of core damage following a rapid boron dilution event are difficult in several areas. First, the size, shape, location, and spatial boron content of the postulated pocket of low boron concentration is speculative. Second, an accurate prediction of the boron transport and turbulent mixing as the pocket is swept to and through the core requires state of the art hydraulics modeling and is dependent upon plant-specific design. We know of only one such analysis (ref 4). Third, determination of the reactivity effect of the spatial boron transient requires three-dimensional analysis. The static 3-D analyses are well within the state of the art, but will vary with each core design and perhaps with each reload. Fourth, a detailed transient 3-D analysis is necessary to reasonably predict local conditions near the hot spot. Point kinetics (or 1-D or 2-D) calculations tend to be either indefensible or extremely conservative. Fifth and last, converting localized fuel conditions to core and RCS damage states requires judgement subject to either uncertainty or conservatism.

The failure threshold for unirradiated and intact fuel rods is 210 to 220 cal/gm radially averaged peak fuel enthalpy (ref 10). In this range, the failure mechanism is brittle fracture of the cladding caused by severe oxidation. This failure mechanism does not lead to fuel dispersal or severe core damage; i.e., there would be no significant fuel dispersal, the fuel rods would remain in a coolable geometry, and no pressure pulses would occur.

When the fuel enthalpy exceeds about 300 cal/gm for unirradiated fuel (equivalent to gross melting of the UO_2 fuel pellet), the fuel failure mode changes. Molten fuel is expelled into water and fragments into small particles. Mechanical energy in pressure waves following fuel dispersal could threaten the integrity of the pressure boundary. The threshold for detecting nuclear to mechanical energy conversion appears to be about 325 cal/gm for unirradiated intact fuel, with a conversion efficiency less than 1% in the range of 325 to 500 cal/gm (ref 10).

The above failure thresholds are lower for irradiated fuel.

Rapid Boron Dilution Scenarios

Various studies have evaluated various boron dilution scenarios, both probabilities and deterministics. In one of the most complete such studies, Sven Jacobson (ref 4) evaluated 15 scenarios and was able to dismiss 12 of them on the basis of very low probability. Evaluation of one event (steam generator tube rupture with backfill during cooldown), although low frequency, led to modification of the Westinghouse tube rupture recovery procedure. The other two events (the so-called French and Swedish scenarios) required further analysis. They are discussed in more detail later.

Six illustrative scenarios for rapid boron dilution, drawn from various sources, are described below.

Scenario A (Dilution During RCS Filling)

The reactor is shutdown and being cooled by the RHR system. The steam generator tubes are drained, but the RCS is in the process of being filled and pressurized. As a result of a CVCS malfunction or operator error, for some period of time diluted water is supplied to the charging pumps, which in turn supply seal injection water to the RCPs. This supply water will be colder than the RCS coolant, and will run into the RCP suction piping and displace the water that is already there. Normal charging is assumed to be isolated. The existence of the clean water pocket is assumed not to be detected by either boron sampling or source range count rate. (This requires either several procedural violations or a "smart" dilution that stops diluting before overflowing from the suction piping into the rest of the RCS.) When the RCP is started in that loop, the clean water pocket is swept into the core. Jacobson estimates the frequency of this event as about 1.E-10/yr - too small for further consideration (ref 4). However, the frequency estimate is quite plant dependent. For example, some plants routinely fill via RCP seal injection (an abnormal occurrence with conditional probability of about 1.E-4 in Jacobson's evaluation). In principle, this scenario could also occur as a result of a deliberate but ill-advised dilution during RCS filling combined with a malfunction of the Chemical and Volume Control System. (Typical U.S. practice calls for boron dilution only after reaching hot zero power conditions.)

Scenario B (Steam Generator Inleakage - The "Swedish Scenario")

The reactor is shutdown and being cooled by the RHR system. As a result of a steam generator tube leak (most likely caused by improperly completed steam generator maintenance or inspection), secondary water enters the reactor coolant system, and collects in the RCP suction piping, steam generator outlet plenum, and perhaps in the steam generator tubes. The existence of this pocket is assumed to be undetectable by normal boron sampling of the reactor coolant being circulated, and assumed to be undetected (although detectable) by mass balances. Subsequent start of the RCP sweeps the clean water into the core. This scenario, sometimes referred to as the "Swedish scenario", is discussed in more detail later in this report. Note that, except for the mechanism of introducing unborated water into the RCP suction, it is identical to Scenario A.

Scenario C (Loss of AC Power during Dilution - The "French Scenario")

The reactor has just been refueled and is the process of being started up. A boron dilution (toward the critical boron concentration) is in progress when a loss of offsite power occurs, resulting in the trip of all RCPs. Decay heat is low and natural circulation does not occur in the reactor coolant loop(s) receiving the diluted charging flow from the Volume Control Tank (VCT). Emergency power comes on and automatically restores the charging flow. In the absence of alarms drawing attention to the dilution in progress, the operators fail to secure the dilution (as required by plant Technical Specifications), and the entire volume of the VCT is discharged into the RCS. The loss of AC power also may automatically isolate letdown, (with the assumption that it is not manually re-established), such that the incoming charging flow is not heated by the regenerative heat exchanger. The incoming unborated and possibly cold water therefore forms a stagnant and relatively unborated volume in the cold leg and bottom of the reactor vessel. When the VCT level is low, charging pump suction is switched to the borated (and cold) Refueling Water Storage Tank (RWST). But before a significant amount of borated RWST water is added to the RCS (borating the unborated volume), offsite power is restored, the operators restart an RCP, and the clean water is swept into the core. This boron dilution scenario is sometimes referred to as the "French Scenario" since it was first publicized in reference 8, and led EdF to install an automatic system to switch the charging pump suction from the VCT to a borated source on loss of power to the reactor coolant pumps. The scenario is also described in references 4, 5, and 11, and is discussed in more detail later.

Scenario D (Unborated Water in RHR System)

Startup of the RHR system when it contains totally unborated water was evaluated by the French (ref 12) with the extreme assumption that no mixing occurred, such that a clean water front moved through the core at a speed determined by the RHR flow; causing the core average boron concentration to drop linearly from 1500 ppm to 0 ppm in 75 seconds. The interesting result revealed by the analysis was that the core power generation caused RHR system rupture on overpressure after about 8 seconds, stopping the dilution well before fuel damage limits were reached. This result suggests that RHR flow is incapable of causing a core disruptive reactivity accident. The estimated frequency of this scenario was on the order of 1.E-8/year, too low for further consideration in any event.

Scenario E (Boration after Shutting Off RCPs)

During RCS cooldown at the beginning of a refueling shutdown, the RCPs are postulated to be turned off at a relatively high temperature (such as 140°F) and before borating to refueling boron concentration. With hot water left in the steam generators, RHR flow would be unlikely to force circulation through the steam generator tubes, with formation of a stagnant pocket of low boron concentration. During subsequent draining and refilling, this pocket would remain in the crossover leg and/or steam generator. Common PWR practice is to continue RCP operation until refueling boron concentration is reached. This practice should be considered mandatory.

Scenario F (Dilution of Refueling Cavity)

A scenario of interest, not involving RCP restart, involves local dilution of the refueling cavity water. Two recent events (ref 13) illustrate the potential for significant local dilution during refueling operations. Specifically, these events involved the formation of a diluted layer of water at the top of the refueling cavity.

In one case, unborated water was used to spray down the reactor internals to minimize airborne contamination. Because the unborated water was less dense than the borated water in the refueling cavity, the unborated water remained near the top of the refueling cavity. Additionally, because cavity water level had to be maintained relatively constant to permit continued maintenance work, the more dense borated water was removed from the bottom via the cavity drain system. After the cavity was refilled in preparation for refueling, the boron concentration in the cavity was sampled, revealing a concentration of 650 ppm near the surface, as opposed to 1950 ppm at the bottom of the refueling cavity. The stagnant layer was about two feet in depth. In the second case, demineralized water was used to spray down the cavity walls as the cavity level was being reduced after refueling. A miscommunication resulted in an excess of demineralized water being used to perform the task. As a result, a diluted layer formed at the top of the refueling cavity, reducing the RCS boron concentration from 2663 ppm to 2626 ppm. This event occurred when the source range detectors were inoperable, preventing the operations staff from detecting the decrease in boron concentration as the diluted layer passed through the active fuel region. With this scenario, the cavity is then postulated to be drained, lowering this diluted layer into the reactor vessel and eventually into the hot leg piping (for mid-loop operation to remove steam generator nozzle dams). The diluted water would then be drawn into the RHR suction, discharged into a cold leg, and forced into the reactor core. (Note, that based on the discussion of Scenario D above, this scenario might cause criticality and significant power generation, but does not appear capable of causing core damage.)

In our ORAM studies, we made detailed evaluation of the French and Swedish scenarios (C and B above), with opposite conclusions.

For the French scenario, we concluded that several significant conservatisms had been made in the references concerning the plant design, expected operator actions, and inherent phenomenology. In particular, we took exception to the assumption that natural circulation would not exist in the loop receiving supposedly unborated charging flow. We performed calculations assuming 0.05% core decay heat (roughly equivalent to decay heat 4 months after shutdown and with replacement of one-third of the core). We found natural circulation flows of about 800 gpm per loop, or roughly 7 to 8 times the postulated charging flow. Perfect mixing of the incoming charging flow with the RCS flow would be expected even if the charging flow had no heating in the regenerative heat exchanger. Mixing would be further assured by: (a) forceful jet impingement of the charging flow stream (about 6 feet/second) and splashing off the opposite wall of the cold leg pipe; (b) warmer charging flow in the highly likely case that letdown flow is re-established; and (c) further mixing of fluid with natural circulation flow from the other loops.

Natural circulation flow in one loop could be stopped, at least temporarily, if auxiliary feedwater flow to other steam generators subcooled them so much that all natural circulation went to them. Hot water in the reactor core outlet will not tend to rise into a steam generator containing even hotter water. Auxiliary feedwater is automatically actuated to all steam generators on loss of offsite power. An early step in typical loss of offsite power recovery procedures instructs the operator to throttle auxiliary feed flow to prevent excessive cooldown. There's some (fairly small) chance that he would terminate auxiliary feed flow to the loop with the charging line much sooner than to the other steam generators. As mentioned, that would be a temporary condition and would not prevent natural circulation and mixing in the reactor vessel. Any temperature difference between the charging flow and the RCS, of course, will cause convection currents within the cold leg between the charging line and the reactor vessel.

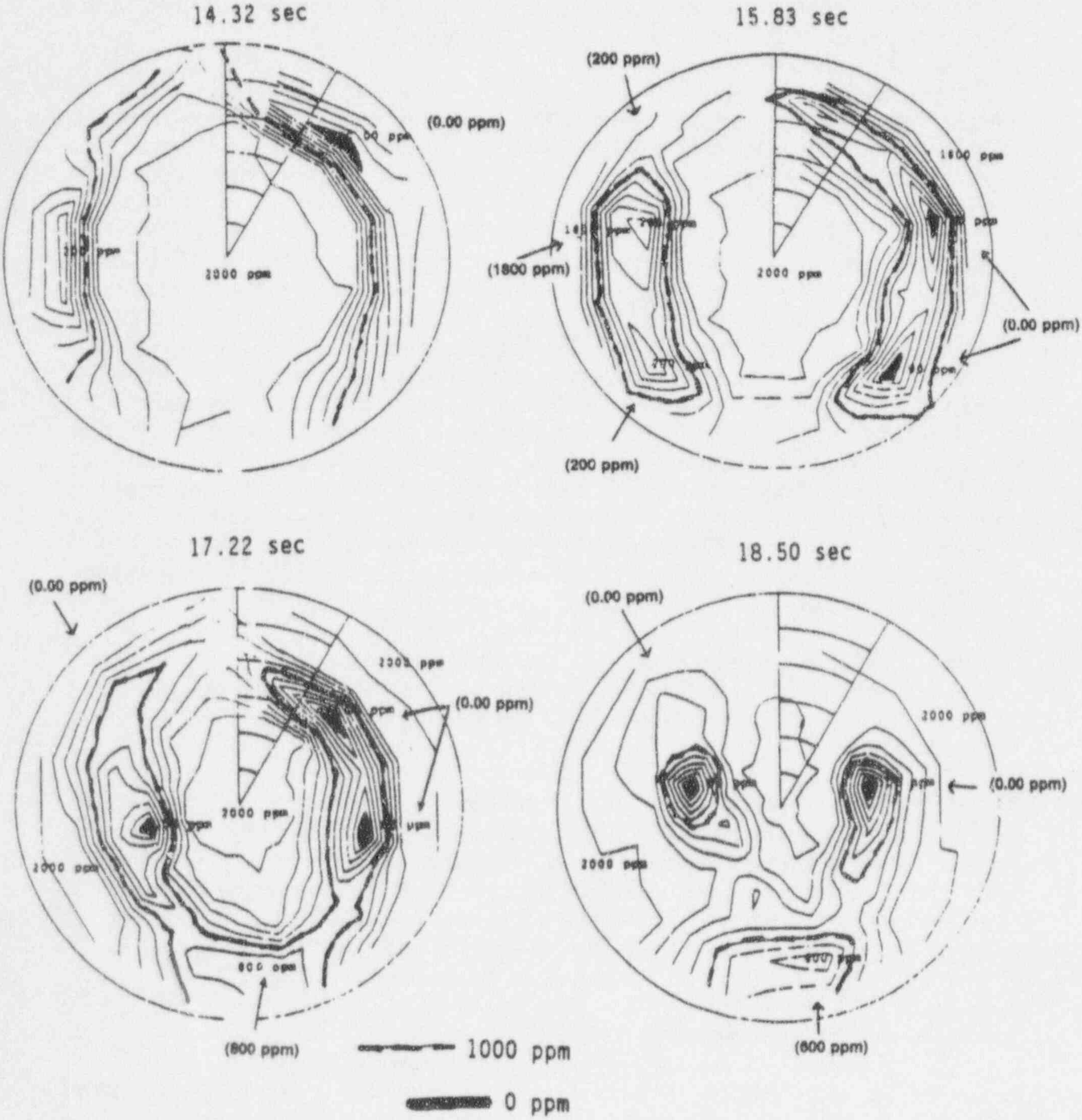
In the Swedish scenario, a large volume of undiluted water is postulated to leak in the RCP suction piping and steam generator from the secondary, and be swept into the core when the RCP is started. Jacobson lists 5 precursor events extracted from Nuclear Power Experience, all during the period 1976 through 1982, in which inadvertent RCS dilution occurred as a result of steam generator maintenance. In addition to those five precursors, secondary water was introduced into the Blayais (French) reactor in March, 1990, through an open steam generator tube (ref 15). Post-event analysis showed that if the operators had not taken action to add boron and stop the dilution, criticality might have occurred in approximately four hours. Based on the precursors he listed, Jacobson concluded that the estimated frequency of occurrence was greater than 1.E-8/yr and required further analysis.

Note that this scenario is identical to Scenario A (Dilution during RCS Filling), and very similar to Scenario E, except for the source of unborated water.

This scenario has been treated extensively only by Jacobson. However, analysis of incorrect startup of an isolated loop has been reported for a VVER plant (six-loop PWR of Soviet design), using a flux synthesis method to approximate the abnormal flux shape (ref 19). Reduction of boron concentration to 600 ppm in one coolant loop resulted in a peak fuel pellet enthalpy of 234 cal/gm, causing limited fuel damage.

For his assessment, Jacobson postulates a small secondary-to-primary leak that takes several days to fill the entire 280 ft^3 (2100 gallons) of the stagnant zone upstream on the RCP. (A larger leak would spill into the active region of the RCS and be detected by boron sampling as well as by mass balances.) Starting the RCP sweeps this water toward the reactor core. Particle tracking analysis with the PHOENIX hydraulics code was used to determine the boron spatial transient into and through the core. The attached figure shows the calculated spatial boron concentration at the core inlet at various times in the transient. Note that considerable breakup of the clean water slug has occurred, yet large variations in boron concentration exist (from 2000 to 0 ppm). The time-varying

Transient Boron Concentration at Core Inlet
for RCP start with Clean Water Pocket
(Jabobson, ref 4, reprinted with permission)



spatial boron concentration was input into a 3-D static nuclear design code (SIMULATE-3) to determine the time varying reactivity. For the reference core design, he calculated a 9% reactivity increase, at rates up to 5% dk/sec (7 \$/sec). This reactivity gain was insufficient to overcome the available shutdown margin (in excess of 15% for that core design), so no criticality occurred. A perturbation case assuming a 500 ft^3 (3700 gallons) clean water pocket indicated a 12% reactivity increase.

Hydraulic tests have been conducted in Sweden to confirm the conservatism of the PHOENIX particle tracking model as used by Jacobson.

These results were calculated for a core with a large shutdown margin - k_{eff} of 0.83 at the normal refueling boron concentration of 2000 ppm (more than 15% shutdown margin). Jacobson noted that extrapolating these results to a more normal core configuration, with a 10% shutdown margin (typical for Ringhals), would give an uncomfortably small margin considering the uncertainties involved.

In the U.S., the industry trend is toward higher enrichment cores with a 5% shutdown margin, for which the extrapolated results would be more severe. Jacobson's results indicate a 10% reactivity gain (reactivity is dk/k), and a change in k_{eff} from 0.83 to 0.92 is equivalent to a change from 0.95 to 1.05. Further, Jacobson's reference core had only a few isolated fresh fuel assemblies, and those on the core periphery, whereas most current U.S. cores are "low leakage", with the most reactive fuel on the inside. The relatively large regions shown on the figure with boron concentrations of 0 to 1000 ppm could well coincide with the most reactive regions of the reactor core. (Note that a core with k_{eff} of 0.95 at 2000 ppm would be critical at about 1600 ppm; and that diffusion and slowing down distances in a cold PWR are short enough that criticality can occur in just a few assemblies.) Finally, cores are being designed that require more than 2000 ppm to obtain a 5% shutdown margin, and the reactivity perturbation is proportional to initial boron concentration.

For these reasons, a straightforward extrapolation of the Jacobson results to a core representative of U.S. trends, suggesting a rapid reactivity transient raising k_{eff} from 0.95 to 1.05 at rates in the neighborhood of \$8 per second, would be both crude and not necessarily bounding.

In our opinion, no amount of analyses of this scenario, using rigor and conservatism appropriate for design basis analysis, will conclusively prove acceptable results for this scenario for all core reload designs. In fact, we strongly suspect that accurate, rigorous analysis will prove the opposite - that unacceptable results (i.e., fuel dispersal) can occur for some cases. Therefore, we conclude that this scenario must be prevented by ensuring that a pocket of unborated water does NOT exist prior to turning on RCPs.

Precautions to Prevent Rapid Boron Dilution

Scenarios with a large volume of unborated water upstream of a Reactor Coolant Pump appear capable of causing severe core damage as a result of a rapid boron dilution. The necessary volume of unborated water appears larger than the crossover leg piping itself, and should therefore be both detectable and preventable. In our opinion, startup of an RCP with a large volume of clean water cannot be proven to be generically acceptable (but cases can probably be found that can be proven to unacceptable). Therefore, we recommend that precautions be taken to ensure a very low probability of occurrence for scenarios A, B, and E. These precautions include:

- o Following steam generator maintenance that affects the integrity of the RCS pressure boundary (such as removing a section of tubing), verify the leak-tightness of the RCS prior to RCS filling and venting operations. This can be done as simply as filling the steam generator secondary and visually inspecting for leaks into the RCS prior to replacing the manways.

- o Don't dilute unless at least one RCP is running;

- o Restrict dilution to conditions under which natural circulation would be expected if the RCPs were tripped; i.e., heat transfer from the RCS to all unisolated steam generators (avoid dilution when one steam generator is hotter than the RCS);

- o When borating for a shutdown, keep at least one RCP running until the desired boron concentration is reached; e.g., to refueling concentration if in a refueling shutdown;

- o If normal filling of the RCS is through the RCP seals (with the normal charging line isolated), review the administrative controls and design to ensure that the probability of inadvertent filling with clean water is vanishingly remote;

- o Boron sampling of the crossover leg of the RCS is recommended prior to RCP restart AND prior to RCS filling whenever the possibility exists that water of reduced boron concentration may exist there. For most PWRs, local samples would have to be drawn from drain lines ("grab samples"). Draining of the crossover leg and refilling with borated water from the active portions of the RCS is an alternative to sampling on PWRs with loop isolation valves. [NOTE: Unexpected readings demand investigation. Since clean water is lighter than borated water, a clean water pocket could exist in the steam generator outlet plenum for a long time without mixing with the more dense borated water at the bottom of the crossover leg where the drain is located.]

Conclusion

A rapid boron dilution event is defined as a relatively unborated volume of water being swept into the core, causing a local reactivity perturbation. One example would be a Reactor Coolant Pump startup with stagnant and unborated water in the RCP suction. Rapid boron in local core boron concentration is a low frequency event - no such case has ever been reported.

In the so-called Swedish scenario, clean water is introduced into the RCP suction during a refueling outage (such as by improper steam generator maintenance), and is swept into the reactor core when the RCPs are jogged as part of filling and venting prior to RCS heatup. Reactivity insertion rates in excess of $10 per second have been calculated for this event, and higher rates appear possible. No recent three-dimensional transient nuclear-thermal-hydraulic analyses involving power generation have been reported, and such analyses are subject to considerable uncertainty, particularly regarding turbulent flow through complex geometry. Existing analyses are inconclusive as to whether severe core damage (expulsion of molten fuel into the core coolant with attendant pressure waves) could result for limiting cases. In our opinion, this scenario cannot be proven to be generically acceptable. Indeed, we believe cases can probably be found that can be proven to have unacceptable results. Therefore, we recommend that precautions be taken to ensure a very low probability of occurrence. These precautions do not appear unduly restrictive, and are generally common U.S. practice.

There have been at least eight precursor events in which some volume of clean water has been put into the RCS as a result of faulty steam generator maintenance (either secondary water through tube leaks or leaks in nozzle dams). None of these events have occurred in the U.S., and all were detected prior to filling and venting. The absence of potential dilution events in the U.S. since 1983 suggests that steam generator maintenance practices have evolved considerably.

In addition to ingress of secondary water, other potential causes of clean water ingress into the RCP suction include inadvertent or ill-advised dilution during RCS filling; malfunction of the Chemical and Volume Control System during filling; injection of cold and unborated water into either the RCP seals or the crossover leg (between the steam generator and the RCP) during shutdown; inadvertent (or "sneak") draining of clean water into the RCS crossover leg during shutdown; and failure to borate to refueling concentration prior to turning off all RCPs. The likelihood of each cause would be strongly dependent upon station practice. Knowledge of the potential hazard, followed by review of station practices and revisions as appropriate, appears to be the most effective way to minimize the likelihood of occurrence.

References

[NOTE: This paper has been extracted from section 3 of NSAC-183, "Risk of PWR Inadvertent Criticality During Shutdown and Refueling", prepared by Westinghouse (Toby Burnett, et al), for the Electric Power Research Institute as part of the EPRI Outage Risk Assessment and Management (ORAM) Program, December 1992]

1. S. Salah, C. E. Rossi, J. M. Geets, "Three-Dimensional Kinetic Analysis of an Asymmetric Boron Dilution in a PWR Core", Westinghouse Nuclear Energy Systems, Transactions of the American Nuclear Society, Vol 15, p. 831, 1972. Also, "Consequences of Asymmetric Cold Water Addition to a PWR Core from an Inactive Loop" by the same authors, Transactions of the American Nuclear Society, Vol 14, p. 755, 1971.

2. Sven Jacobson, "Possibility to Get Prompt Criticality in Conjunction with RCP Start", Swedish State Power Board letter to Westinghouse/Brussels, R4-SVWE-739, December 1982.

3. NUREG-1251, "Implications of the Accident at Chernobyl for Safety Regulation of Commercial Nuclear Power Plants in the United States", April, 1989.

4. Sven Jacobson, "Risk Evaluation of Local Dilution Transients in a Pressurized Water Reactor", Linkoping University, Sweden, March 1992.

5. NUREG/CR-5819, "Probability and Consequences of a Rapid Boron Dilution in a PWR: A Scoping Study", D. J. Diamond, et al, Brookhaven National Laboratory, U.S. NRC, June, 1992.

6. NUREG/CR-5368 (BNL-NUREG-52198), "Reactivity Accidents", January, 1990.

7. Sven Jacobson, "Some Local Dilution Transients in a Pressurized Water Reactor", LIU-TEK-LIC-1989:11, Linkoping University, Sweden, 1989.

8. EDF Report, "Probabilistic Safety Assessment of Reactor Unit 3 in the Paluel Nuclear Power Centre", Moroni, et al, May, 1990.

9. NSAC-176, "Safety Assessment of PWR Risk During Shutdown Operation", Toby Burnett, et al, Westinghouse Electric Corp., Electric Power Research Institute, June, 1992.

10. T. Fujishiro, "Current Status of Research and Regulation Relevant to Reactivity Initiated Accidents in Japan", Japan Atomic Energy Research Institute report to the Technical Committee Meeting on Reactivity Transient Accidents, IAEA, Vienna, Austria, 17-20 Nov 1987.

11. NRC Information Notice 91-54, "Foreign Experience Regarding Boron Dilution", Sept 6, 1991.

12. A. Chesnel, "Analyses of Reactivity Accidents in PWRs", Commissariat a l'energia atomique (France), report to the Second Technical Committee Meeting on Safety Aspects of Reactivity Initiated Accidents, IAEA, Vienna, Austria, 14-17 Nov 1989.

13. INPO Significant Event Report 18-89, "Potential for Reduction in Boron Concentration in the Reactor During Refueling Activities", August, 1989.

14. H. Nourbakhsh, BNL, personal communication, August 20, 1992.

15. INPO Significant Event Report 13-90, "Unplanned Boron Dilutions from Shutdown Conditions, August, 1990.

16. G. Alloggio, et al, "Low Power Reactivity Initiated Accidents in New Generation

Light Water Reactors", ENEL report at the IAEA Technical Committee Meeting on Progress in Development and Design Aspects of Advanced Water-Cooled Reactors, Rome, Italy, September, 1991.

17. Zion Nuclear Generating Station, GOP-1, "Plant Heatup".

18. Zion Nuclear Generating Station, MI-2, "Reactor Coolant System Fill and Vent".

19. M. Antila, et al, "Application of the Synthesis Model in an Asymmetric Reactivity Disturbance of the VVER-440 type Lovissa Reactors", ANS Topical Meeting on Safety of Thermal Reactors, Portland, OR, July, 1991.

20. Sven Jacobson, personal communication, Aug 27, 1992.

BORON DILUTION REACTIVITY TRANSIENTS, AN OVERVIEW OF EFFORTS.

By Anthony C. Attard
Office of Nuclear Reactor Regulation
United States Nuclear Regulatory Commission

Specialist Meeting on Boron Dilution Reactivity Transients
State College, PA, U.S.A., October 18th-20th, 1995
SESSION I: OVERVIEW OF PAST AND PRESENT EFFORTS

BACKGROUND

The U.S. Nuclear Regulatory Commission (NRC) staff has reviewed several reports regarding various boron dilution scenarios that could lead to an inadvertent core recriticality. The subject of the reports ranged from the accumulation of highly diluted water due to loss of offsite power during startup (the so-called "French scenario"), to reflux condens ion following a small-break loss-of-coolant accident (SBLOCA). (References 1 through 7).

The events involve the accumulation of a significant volume of unborated water in a part of the primary coolant system during plant shutdown or following an accident. Various actions performed during plant shutdown could result in such an accumulation. Once the unborated water is accumulated, the start up of an idle reactor coolant pump could send the unborated water into the reactor core causing a significant and unplanned reactivity insertion.

The analyses indicate that there is a potential for a rapid injection into a PWR reactor core of unborated reactor coolant whose effects would be more severe than similar events considered in the safety analyses of most U.S. plants.

The accident scenarios described in the referenced studies involve various assumptions regarding plant conditions and equipment configuration and therefore, may not apply to all, or any particular plant. However, it is clear that training and procedures that emphasize the need at all times to ensure uniform boron concentration in the reactor coolant system, and the implementation of appropriate actions and cautions in starting an idle reactor coolant pump, can reduce the probability of occurrence of such an event.

Following the review of foreign and in-house studies, the NRC responded by issuing an information notice `1-54 (Ref. 8), pertaining to "Foreign Experience regarding Boron Dilution." The information notice alerted the addresses to a potential sequence of events that may result in a rapid injection of unborated coolant water into the reactor core.

In addition, the NRC staff has reviewed other studies and related technical papers covering several deboration scenarios. The staff also discussed the subject of deboration with reactor vendors and scientists from some national laboratories where research in boron dilution in the shutdown mode and, reflux condensation have been conducted

ONGOING LICENSING ACTIONS

Boron Dilution During SBLOCAs

The staff asked ABB-CE to address the applicability of this boron dilution event to the System 80+ design and to resolve the issue.

In response to the staff's request (Ref. 9), ABB-CE submitted the results of their evaluation of the potential for RCS boron dilution during an SBLOCA. Basically, the postulated SBLOCA scenario results in the accumulation of deborated water in each of the RCS cold-leg loop seals. The mechanism for accumulating deborated water in the loop seals is caused by steam condensation (reflux cooling) following drainage from steam generator (SG) tubes. During reflux cooling, the condensate on the cold-leg side of the SG tubes drains into the loop seals. The staff was concerned that in this configuration, the introduction of deborated water in the core would have deleterious effects on maintaining subcriticality. ABB-CE stated that low boron concentration in the

System 80+ loop seals may occur for small break sizes between 2.54 and 7.62 cm (1 and 3 in.) in diameter.

A bounding analysis was performed without crediting any of the mixing of borated and unborated water which is expected to occur in the RCS. Instead, the condensate was assumed to enter the core as an unlimited size slug of pure water moving at a natural circulation flow rate consistent with that of a small break at the time of RCS refill using ECCS injection.

A core physics analysis was performed to determine the reactivity, the power peaking, the power transient and the minimum critical boron concentration required to avoid recriticality at beginning of the cycle with all rods inserted as the unborated slug passed through the core. An RCS thermal-hydraulics analysis also was performed to determine the change in pressure and in natural circulation flow rate as energy from the core entered the coolant.

The analysis indicates that the core returned to a critical condition when the unborated slug progressed partly through the core. As the slug progressed further into the core, the resultant neutron power function experienced a very brief spike, which was terminated by Doppler feedback in the fuel. The power then dropped further as coolant heatup resulted in moderator density reactivity feedback. The power underwent several oscillations of diminishing amplitude and finally settled at a level that was a small fraction of full power. The analysis indicated that boron concentration required to avoid a return to criticality at beginning of the cycle with all rods inserted, depends on the temperature of the coolant. An average boron concentration of about 550 ppm is required to avoid a return to criticality at 149 °C (300 °F), but only 200 ppm is required at 260 °C (500 °F).

ABB-CE concluded that if the analysis had accounted for borated water entering the core behind the slug, then the power would have rapidly decreased to zero. The staff finds this acceptable because during RCS refill, the SG tubes will fill through the hot leg, and the borated ECCS water (4400 ppm) would flow and mix with the deborated loop seal before entering the reactor vessel. Once in the vessel, the staff believes that additional mixing would occur through the downcomer.

For natural circulation conditions, ABB-CE calculated that the time required for the condensate in the loop seals to pass through the core is approximately 3.3 minutes and it will take two to three times longer for the condensate to pass through the RCS. ABB-CE also postulated that the consequences for restarting the RCP wouldn't be of concern if procedural restrictions delayed the restart process for at least 20 minutes under natural circulation conditions.

The staff agreed that a 20-minute delay is a conservative time limit to permit the condensate to pass through the RCS at the natural circulation flow rate (approximately 2 percent to 3 percent of total flow) and mix with the highly borated coolant in the RCS. However, the staff raised concerns that the operator could err in determining that natural circulation is established, and for how long it is established. Because of the potentially serious consequences of an operator prematurely restarting an RCP (assuming the presence of an unborated slug), the staff suggested that procedural controls alone may not be adequate. Consequently, ABB-CE was requested to demonstrate that the event is incredible; the consequences are not serious; or provide additional protective measures.

<u>SBLOCA Deboration Events With Restarting an RCP</u>

In response to the staff's concerns, ABB-CE submitted their analysis and changes to Emergency Operations Guidelines (EOGs) described in CESSAR-DC, Appendix 6C to support their position that the SBLOCA deboration event with restart of an RCP is unlikely to occur, and that even if an RCP restart was to occur under this condition, the consequences are not serious. The staff's review of CESSAR-DC, Appendix 6C and proposed EOG changes is provided in the following evaluation.

Background

The ABB-CE system response analysis showed the potential for an SBLOCA deboration event for SBLOCA break sizes ranging from 2.54 to 7.62 cm (1 to 3 in.) in diameter. These sizes of break are small enough that the break flow is not sufficient to remove all of the decay heat. The secondary side will be relied on for the decay heat removal. Also, these break sizes are large enough so that the break flow is greater than the safety injection flow, thus reducing the RCS water level below the bottom of the cold leg. At this water level, the steam generated in the reactor core will be transferred to the tube side of the steam generator (SG). When RCS cooling by the secondary side is initiated, the reflux/condensation process begins.

Formation of the condensate will result in some of the condensate flowing back via the hot legs to the reactor vessel, counter-current to the steam flow. The rest of the condensate will flow into the loop seals (RCS suction pipes) and collects there until the RCS refills and natural circulation is regained. If the operator inadvertently turns on the RCPs, the unborated (or low borated water) in the loop seals could be transported into the core rapidly, and cause a rapid reactivity transient.

ABB-CE performed a probabilistic risk assessment (PRA) of the SBLOCA deboration event for System 80+. Important factors in the analysis included: the likelihood of an SBLOCA, the amount of boron mixing in the cold-leg piping during the refill phase of the event, reestablishment of natural circulation in the primary system, and the likelihood of an operator restarting an RCP prior to the establishment of natural circulation.

Small break LOCA frequency is typically estimated to be in the range of 10^{-2} to 10^{-3} per reactor year, and is therefore a potentially significant initiator. Mixing in the cold-leg and the loop seal may occur during the refill phase as a result of highly borated water from the SI pumps flowing back through the RCP into the loop seal. Low rates of natural circulation will resume once the lower tubes of the steam generator are filled. The natural circulation rate (and mixing process) will increase until the system is refilled. ABB-CE's analysis indicates that when all steam generator tubes have filled, the natural circulation rates result in a well mixed primary system in about 20 minutes. ABB-CE also analyzed the reactivity effect of unborated water entering the core at the natural circulation flow rate.

In response to the staff's concern, ABB-CE changed the System 80+ EOGs to ensure that the operator will not inadvertently turn on the RCPs during an SBLOCA event. The EOGs changes are as follows:

(1) The RCP Restart Strategy

The RCP restart strategy mainly involves five steps. Since maintenance of natural circulation (NC) prior to restart of the RCP will help operators avoid unacceptable core conditions from occurring during an SBLOCA event, the RCP restart steps were modified in the following priority in order to emphasize the importance of maintenance of NC before turning on an RCP:

(a) Verify adequate single-phase NC

(b) If single-phase NC cannot be established, verify adequate two-phase NC

(c) Determine if RCP restart is needed and desired

(d) Verify that all RCP restart criteria are met

(e) Restart RCPs.

These modifications are reflected in the LOCA Functional Recovery Guidelines.

(2) RCP Restart Desirability and Criteria

Step (1)(c) above provides guidance for the RCP restart desirability and Step (1)(d) provides acceptance criteria for RCP restart. To further ensure that the operator will not inadvertently restart the RCP prior to establishment of NC, two modifications were made to each of these steps: one modification requires the operator to obtain concurrence from the technical support center (TSC) on the RCP restart; the other requires the operator and the TSC to consider the length of time that the plant had been in NC when evaluating the desirability of RCP restart.

(3) Supplementary Information Item

It is important for the operator to consider whether or not deborated water could build up in the suction leg of the RCP prior to RCP restart. ABB-CE has added supplementary information to the LOCA recovery guideline that cautions the operator about this possibility prior to RCP restart. In addition, ABB-CE has specified that the supplementary information should become a caution in the plant specific procedures. Specifically, this caution is intended to be placed prior to the step for RCP restart desirability determination (Step (1)(c) above) in the plant specific procedures.

(4) Modifications to the EOG Bases

The Bases section was modified to match the new step order and to contain the bases explanations for the new steps.

The staff has reviewed these modifications to the EOGs. Since the modified EOGs require the operator to take many steps to restart an RCP, including checking with the TSC to confirm that natural circulation has commenced, thus assuring adequate shutdown margin, the staff concluded that the modified EOGs provide a reasonable assurance that the operator will not inadvertently restart the RCP during an SBLOCA event.

Boron Mixing Analysis

In assessing the need for modifications beyond the EOG revisions discussed above, the staff also considered the possibility that the operator could err in determining that natural circulation is established and for how long it has been established. ABB-CE was asked to assess the efficacy of boron mixing in the RCS assuming the presence of a large unborated slug and an operator action to start the RCP in the same loop.

As initial conditions for the analysis, ABB-CE assumed that the loop seal and the cold leg volume (below centerline) is filled with pure (unborated) water. This is a volume of 7.42 m^3 (262 ft^3). The computational fluid dynamics (CFD) code (FLUENT) was then used to assess the mixing in the downcomer and the lower plenum of the core following the startup of one reactor coolant pump (RCP). FLUENT is a thermal hydraulic code widely used in the industry to model fluid flow in pipes. It has two dimensional and three dimensional calculational capability for steady or transient flow calculations, compressible or incompressible flow modeling, and can track chemical species distributions in the fluent. Although the staff did not review the details of the code, the staff did review the various required inputs and the applicability of the code to the boron dilution problem

ABB-CE included the following conservatisms in its analyses:

(1) No credit was taken for mixing in the RCP discharge pipe.

(2 No credit was taken for flow entrainment in the reactor vessel.

(3) No credit was taken for mixing at the reactor vessel inlet nozzle as the slug of water hits the reactor vessel plenum wall.

(4) Mixing in the reactor vessel lower head and lower support structure was assumed. This mixing was underestimated by the use of a simplified flow path which ignored the tortuous path the fluid must take in the lower

head and lower support structure area.

The staff concluded that the ABB-CE analysis used conservative assumptions to assess boron mixing effects in the RCS for the SBLOCA deboration event.

The output of the FLUENT code included boron concentration as a function of space and time. The results of the ABB-CE analysis show that the boron concentration of the water entering the core region is heavily dependent upon the initial slug volume. Doubling the initial slug volume (from 7.42 m^3 (262 ft^3) to 15.35 m^3 (542 ft^3)) reduced the minimum boron concentration by approximately 35 percent. A slug size of 15.35 m^3 (524 ft^3) reduced the calculated boron concentration in the core region to 1,350 ppm.

Critical boron concentration is the boron concentration above which criticality will not occur. The critical boron concentrations are also a function of temperature. Analysis by the ABB-CE shows that, at BOC and all rods in (ARI), at 260 °C (500 °F), the critical boron concentration is 200 ppm. At 149 °C (300 °F), the critical boron concentration is 550 ppm. These values are well below the 1,350 ppm obtained as assuming the initial presence of a large slug of 15.35 m^3 (524 ft^3). In addition, ABB-CE shows by neutronic analysis that the rapid reactivity transient may cause recriticality only during the first third of the fuel cycle since beyond this cycle time no boron is required to maintain the core subcritical for post-LOCA conditions with all control rods inserted. Therefore, the staff concluded that ABB-CE's analysis provides a reasonable assurance that sufficient boron mixing can be expected to prevent core recriticality from occurring during an SBLOCA deboration event in the unlikely event that an operator restarts an RCP before natural circulation is fully established.

Conclusion for SBLOCA Deboration Events With Restarting an RCP

The staff has reviewed the analysis and EOGs changes described in CESSAR-DC, Appendix 6C. As a result, the staff concludes that there is a reasonable assurance that the postulated deboration transient during an SBLOCA with an

RCP restart poses no undue threat to the public health and safety.

AP600 Boron Dilution Analysis

Westinghouse is presently in the process of preparing their response to the staff's request regarding deboration scenarios associated with the AP600 reactor. The staff recognize the fact that although the AP600 does not have loopseals per se, accumulation of deborated coolant is feasible in other parts of coolant system. Westinghouse will submit to the staff analyses encompassing all anticipated scenarios.

CONCLUSION

The NRC staff reviewed several studies concerning rapid boron dilution during shutdown and SBLOCA events. Probabilistic analysis indicates that a rapid boron dilution event has a low frequency of occurrence. (Refs. 1 & 2). However, several potential mechanisms could lead to such an event.

No three-dimensional (nuclear and hydraulic) analyses have been performed in recent years; also, analyses such as those reported here, are subject to considerable uncertainty, particularly regarding mixing. Consequently, these analyses must be viewed as conservative analyses.

The staff believes that the events reported can be prevented by the use of appropriate procedures which anticipate the possibility of dilution in various recognized situations and prevent it, or prevent the inappropriate starting of pumps until suitable mixing procedures are carried out.

REFERENCES

1. U.S. Nuclear Regulatory Commission, "Implications of the Accident at Chernobyl for Safety Regulation of Commercial Nuclear Power Plants in the United States," NUREG-1251, Vol. 1, April 1989.

2. U.S. Nuclear Regulatory Commission, "Reactivity Accidents," NUREG/CR-5368, January 1990.

3. U.S. Nuclear Regulatory Commission, "Probability and Consequences of Rapid Boron Dilution in a PWR: A Scoping Study," NUREG/CR-5819, June 1992.

4. Sven Jacobson, "Risk Evaluation of Local Dilution Transients in a Pressurized Water Reactor," Ph.D. Dissertation Linkoping University, Sweden, March 1992.

5. Juhanni Hyvarinen, "An Inherent Boron Dilution Mechanism in Pressurized Water Reactors," Finnish Centre for Radiation and Nuclear Safety (STUK), September 1992.

6. Westinghouse Electric Corporation, "Risk of PWR Inadvertent Criticality During Shutdown and Refueling," NSAC/183, EPRI Outage Risk Assessment and Management (ORAM) Program, December 1992.

7. U.S. Nuclear Regulatory Commission, Certification of System 80+, NUREG-1462, Vol. 2, August 1994.

8. NRC Information Notice 91-54: Foreign Experience regarding Boron Dilution, September 6, 1991.

9. CE-System 80+, Vol. U.S. Nuclear Regulatory Commission, Office of Nuclear Regulatory Regulation, August 1994.

BORON DILUTION EVENTS

FIRST INVESTIGATIONS AND SAFETY VIEW

N. TRICOT - IPSN/DES (*)

(*) IPSN/DES - Institut de Protection et de Sûreté Nucléaire / Département d'Evaluation de Sûreté

BP n° 6 - 92265 Fontenay-aux-Roses Cedex

ABSTRACT

On the occasion of the post-Chernobyl studies performed in France, the importance of some dilution sequences, notably by pure water slugs, was pointed out, and lead as early as 1990 to the implementation of additional complementary provisional measures on all French PWRs. In the mean time, a complementary research and development program has been set up by the utility, in order to define and to implement at the request of the French safety authority a definitive modification, allowing a significant reduction of the risk issued from scenarios with rapid reactivity insertion resulting from event sequences leading to an introduction of a diluted water slug in the core. Furthermore, an identification as exhaustive as possible of potential initiators was requested.

The scope of this paper is to present the main sequences identified at that time, the first probabilistic safety studies results and the provisional measures implemented on the French PWR at the beginning of the 90's.

I - INTRODUCTION

In the frame of the post-Chernobyl studies, started up in 1986, studies have been performed in France by the IPSN (Institute for Protection and Nuclear Safety) and the utility (Electricité de France) on reactivity initiated accidents (RIA). Depending of the initial reactor state, these accidents could not only lead to a severe core damage but also to the loss of the containment function.

Accident sequences have been studied starting from each reactivity initiating event already considered in the safety report, cumulated with human errors and/or system failures and/or technical specification transgressions, until severe core damage occurs.

On the basis of these studies, probabilistic safety studies were performed on the 900 MWe and the 1300 MWe plants, which pointed out in 1990, on the basis of an exhaustive screening of the sequences issued from those initiators, the importance of the risk associated to boron dilution sequences. These accidents can be divided into two types : the progressive dilution of the Reactor Cooling System (RCS) and the fast injection of a slug of diluted water in the core. We will focus hereafter on this last type of dilution mode.

II - MAIN BORON DILUTION SEQUENCES IDENTIFIED IN 1990

Among all boron dilution scenarios, physical studies pointed out that a diluted water slug could lead to a severe core damage in a very short time. Therefore, special care has been given to investigate the situations which could lead to the formation of a diluted water slug in the primary circuit or the connected systems.

On the basis of this first screening, two main sequences were identified which could lead with an important calculated probability to a severe core damage after a rapid sweeping of the core by a slug of pure water ; they are presented hereafter.

II.1 - Loss of the primary pumps during a dilution operation

II.1.1 - Initiating event

This sequence could occur during a normal dilution operation (30 m^3/h), performed to compensate the loss of reactivity due to the fuel burn-up. The loss of off-site power supply during this operation would induce the drop of the control rods and the reactor coolant pumps (RCP) trip. Since the chemical and volume control system (CVCS) is backed-up by an auxiliary power supply, the dilution would go on in a stagnant loop if the natural circulation flowrate is not sufficient to ensure a correct mixing of the injected water. This would result in the creation of a non borated water slug in the primary system.

In such a situation, the operator would have to apply the loss of off-site power procedure which requires to :

- stop the dilution if it is on progress,

- start up a RCP after shifting its power supply to the auxiliary transformer, the corresponding loop being possibly the loop on which the charging line is connected.

The delay before application of the procedure (about 10 mn) might be long enough to accumulate a sufficient volume of pure water in the loop to lead to a reactivity accident when this water is injected in the core following the RCP starting up.

II.1.2 - Physical studies

The preliminary and analytical neutronic studies presented by the utility pointed out that the critical size of the pure water slug entering the core (disk shaped slug hypothesis) and leading to prompt criticality, is approximately one cubic meter (without taking into account any mixing) at the beginning of a fuel cycle. Taking into account the flow dispersion in the downcomer, an equivalent volume of 2.5 m^3 stored in the cold leg was estimated to be sufficient to lead to the critical size in the core.

II.1.3 - Probabilistic safety studies results

The probability of the dilution sequence by a pure water slug as described in § II. 1 was very roughly evaluated in 1990 to 1.2 E-3 / r.y for the 900 MWe and to 2.9 E-3 /r.y for the 1300 MWe reactors (considering that the operator would act within a delay greater than 10 mn). According to the importance of these values, EdF decided to implement on all French PWR plants, additional complementary provisional measures characterised by:

- an automatic switching of the CVCS suction pumps from CVCS to the refuelling and water storage tank (RWST with a boron concentration of 2000 ppm), in the case of a loss of the primary flow with dilution in progress. The corresponding signal also triggers a reset of the dilution sequence (see attached figure),

- adapted administrative lock out procedures.

The above additional measures reduce the calculated probability of core damage associated to this sequence to 2.2 E-7 / r.y for the 900 MWe and 3.5 E-6/ r.y for the 1300 MWe without taking into account any operator action. These modifications were implemented on all French PWR plants before the end of 1991.

II.2 - Start-up of a diluted residual heat removal system (RHRS)

II.2.1 - Initiating event

In normal operation, at full power, the RHRS is kept full of water by connection with the RWST on the 1300 MWe and by the chemical and volume control system (CVCS) on the 900 MWe. In case of leakage on the RHRS, this one will be filled up with water from its makeup source. In this case, on the 900 MWe series, the RHRS could be diluted when the boron concentration in the RCS is almost zero (core end of life). In the case of a start-up of the RHRS, its boron concentration would be lower than the one required by the technical specifications in the cold shutdown state (the boron concentration was evaluated at 600 ppm lower than the boron concentration required in cold shutdown).

The boron concentration in the RHRS must be verified prior to each RHRS start-up. So, a start up with diluted water in the RHRS assumes an error from the operating team. Furthermore, all the RCPs must have been stopped (abnormal situation) prior to the start-up of the diluted RHRS to lead to a criticality accident. If at least one RCP is running, the transient would lead by mixing to only a slight decrease of the average RCS boron concentration with no effect on the fuel.

In the frame of the post-Chernobyl studies, two extreme sets of assumptions were assessed by the utility. The first assumed that the core would be swept by a pure water slug with a flowrate of 1050 m^3/h ; the reactivity insertion rate is then about 250 pcm/s. With this assumption, the rupture of the RHRS by overpressurisation would occur before severe core damage and the accident would evolve toward a loss of coolant accident (LOCA), with a short delay before core melt in case of deficiencies such as errors in accident management or component failure (safety injection). The second set assumed a perfect mixing with the RCS water ; this assumption leads to a reactivity insertion rate of 50 pcm/s and no significant effect on the core.

II.2.2 - Probabilistic safety studies results

The probability of RHRS break was estimated to 1.2 10-6/ r.y for the 900 MWe units and no particular provision was taken for this sequence at that time. However the corresponding core melt probability has still to be evaluated.

As a conclusion, the start-up of a diluted RHRS appears potentially less critical than the loss of the primary pumps during a dilution operation in terms of reactivity injection and probability but could lead to a primary loss of coolant by RHRS rupture by overpressurisation ; at the present time, no specific measures have been thought necessary to prevent this sequence.

III - IDENTIFICATION OF OTHER DILUTION SEQUENCES

The efficiency of the provisional modification is limited to the main identified sequence described in section II. 1. IPSN and EdF independently performed in 1990 a second screening to identify other possibilities of a pure water slug formation, not prevented by the above mentioned modification. From those complementary investigations, the most important sequences pointed out were :

III.1 - During normal operation

- Pure water from reactor boron and water make up system (RBWMS) to CVCS make up, due to human error (not covered by the automatism). This sequence covers notably :

  the boration in hot shutdown state : to ensure an appropriate shutdown margin allowing to join the cold shutdown state, in case of a loss of external power supply for example, the operating procedures require a boration of the primary circuit in hot shutdown state. A human error in the selection of the make up mode (dilution instead of boration), would imply a possibility of boron dilution in the primary circuit. Nevertheless, the residual heat remain sufficient to ensure the homogeneisation of the fluid and the formation of a diluted water slug can be excluded.

  the cooling to join the cold shutdown state : in case of cooling to join the cold shutdown state before opening of the reactor vessel, the operator has to continue the previous action (boration in hot shutdown). For this case, a dilution mode selection seems unlikely, but the operator could use by error the automatic make up, with an inadequate low borated water flowrate. The start-up of a primary pump after a loss of off-site power could lead to a diluted water slug injection into the core. The probability of this scenario of low borated water slug formation was estimated to some 1.1 10^{-6} / r.y for the 1300 MWe plants and 2.5 10^{-6} / r.y for the 900 MWe plants.

  In the case of cooling to the cold shutdown state without opening of the reactor vessel, the primary fluid contraction is compensated by the automatic make up. The initiator of this sequence would be a loss of off-site power concomitant with a low or non borated water make-up ; the corresponding probability was estimated to 1.5 10^{-6} / r.y for the 1300 MWe plants and 1.7 10^{-6} / r.y for the 900 MWe plants.

- Cold slug through reactor coolant pumps seal injection : when the RCPs are stopped in the case of a loss of offsite power, the water injected through the seal injection falls in the seal loop. If the natural circulation is not strong enough, the cold water could be blocked by density effects and a cold water slug could be formed ; the probability of this initiator was estimated in the order of 10^{-4} / r.y.

- Demineralizer resin replacement or start up after resin replacement. A resin replacement can be done in all the reactor states ; for this operation the injection of demineralized water downstream from the demineralizer is needed. In the case of an human error (non isolation of the demineralizer) concomitant with the non closing of the demineralized water distribution system valve, the formation of a pure water slug could occur. The probability of this initiator, if this operation has been done when the primary pumps are tripped, was estimated to some 10^{-6} / r.y (900 and 1300 MWe plants). On another hand, the start-up of the demineralizer without closing of the demineralized water distribution system valve due to an human error during a loss of off-site power was estimated to some 10^{-5} / r.y for the 900 MWe and 1300 MWe.

The reassessment of all these sequences (probabilities / consequences) was required by the French safety authority, taking into account the last research and development results and in a more complete way the operating practices and experience feedback. Furthermore, EdF was requested to propose, if necessary, additional provisions to cope with the remaining sequences.

III.2 - <u>For incidental situations (notably leakages of cooling systems connected to the primary circuit)</u>

- Thermal barrier leakage of a reactor coolant pump,

- Rupture of an heat exchanger,

- Plug mispositionning on a steam generator tube (this scenario actually occurred on Unit 4 of Le Blayais Power plant on March 3th 1990 (described in IRS n°NEA 1111, July 6th 1990), due to human errors resulting in the filling of the steam generator 2 without the plugging of a cut tube. There was an introduction of pure water into the primary system which could only be stopped by draining the affected steam generator. However, the minimum boron concentration requirement of the technical specifications (2000 ppm) was always complied with),

- Steam generator tube rupture : according to the emergency operating procedures, a depressurisation of the primary system could lead to an introduction of unborated water from the secondary side,

- Dilution by the safety injection system :

  Boron concentration too low in the RWST : this scenario can be excluded due to the volume of the tank and the delay which would be necessary to lead to its dilution without any detection,

  Dilution of recirculated water by filling the sumps with pure water : this scenario could occur after a secondary circuit break leading to the containment spray system start-up and the dilution of the sumps,

- Post accidental phase (reflux - condenser cooling mode) : in the case of a small break for example, condensation on the primary side of the steam generators could lead to a pure water slug formation in the intermediate legs.

Complementary studies are necessary to better assess the likelihood of these sequences. Furthermore, the French safety authority requested EdF to propose provisions to avoid the introduction of pure water from the steam generators and the component cooling system.

IV - SAFETY REQUIREMENTS FOR THE ELABORATION OF A DEFINITIVE MODIFICATION

With the intention of elaborating a definitive modification, a research program has been undertaken by EdF (1991) to improve the assessment of the reactivity insertion possibilities. This program includes:

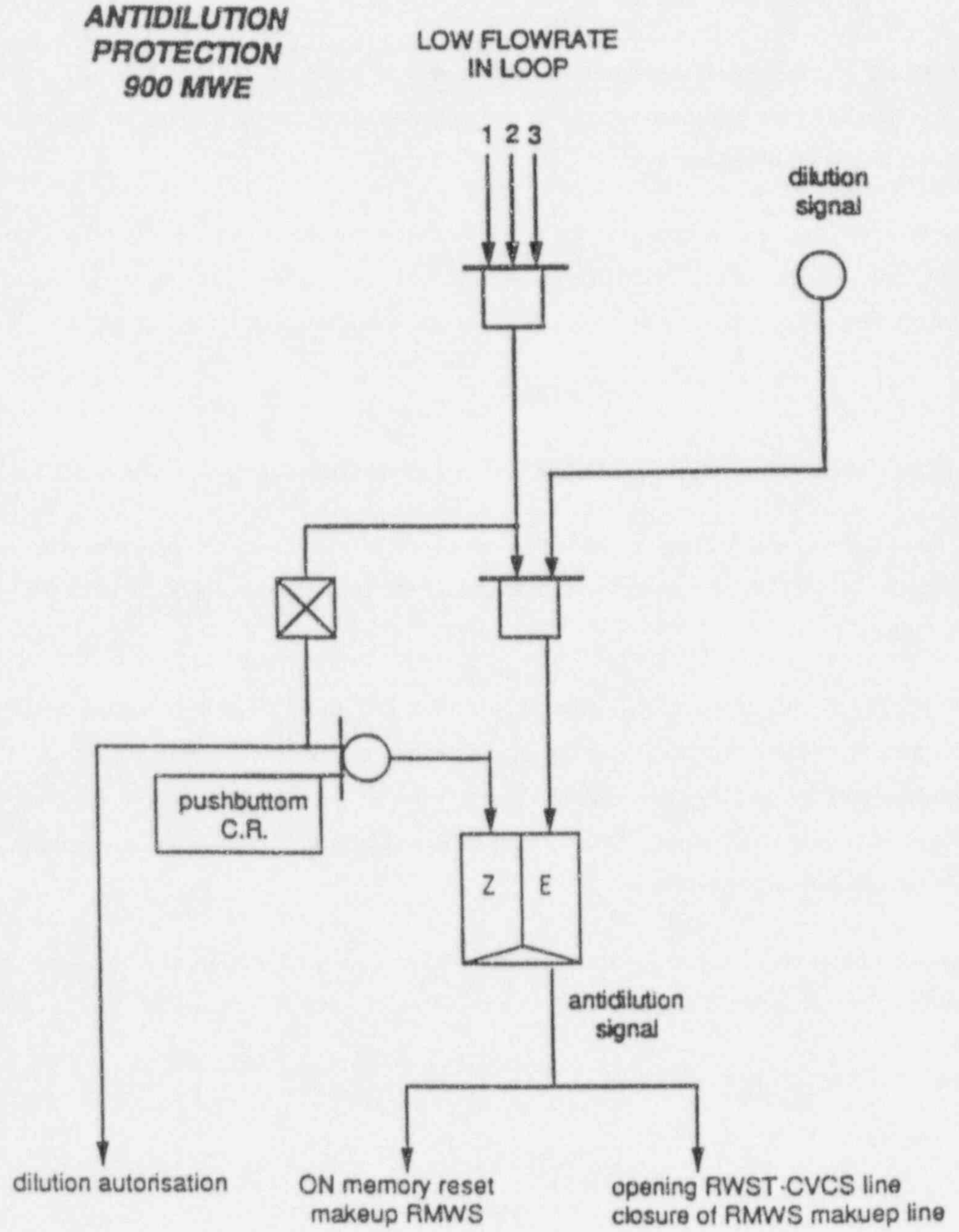
- physical studies, based on neutronic calculations and 3-D thermohydraulic calculations and tests on an experimental thermohydraulic facility. The aim of these studies is to get a better assessment concerning, notably, the critical size of a water slug in the core leading to core damage (see § II.1) and the critical size of a water slug in the cold and / or intermediate leg, taking into account the mixing efficiency,

- a thorough identification of the sequences which could lead to an injection of unborated water or diluted boron solution in the core, including shutdown states and maintenance,

- the evaluation of the corresponding probabilities.

The research p.ogram is now in progress and will be presented by EdF during this meeting as also the EdF proposal for a definitive plant modification. These topics will be submitted in the near future to the French safety authority and assessed by IPSN.

For its part, IPSN has undertaken from as early as 1991 independent studies based on 3D thermohydraulic and neutronic calculations allowing a better understanding of the physical phenomena ; these studies are still in progress and will not be presented in this meeting.



OECD/ NEA/ CSNI Specialist Meeting on
Boron Dilution Reactivity Transients
State College, Pennsylvania, U.S.A., 18-20 October 1995

OVERVIEW TO BORON DILUTION PROBLEM IN LOVIISA NPP

S.Savolainen
Imatran Voima Oy, Loviisa Power Plant, Finland

ABSTRACT

During the last five years boron dilution events have become one of the most interesting topics of PWR safety. The dilution events can be divided into two different mechanisms.: external boron dilution and inherent boron dilution. The main emphasis in boron dilution analyses have been in external dilution events, so far.

In safety analyses for FSAR(Final Safety Analyses Report) and PSA for Loviisa NPP it was turned out that external dilution forms risk for Loviisa plant. That lead to immediate plant modifications. In 1994 Final plant modifications was implemented against external dilution events. In 1993 the Finnish Centre for Radiation and Nuclear Safety (STUK) required Imatran Voima Oy to carry out comprehensive boron dilution studiy. It was foreseen that boron dilution of external sources is not the only potential mechanism.

The study of other potential mechanisms leading to primary circuit dilution was launched in the beginning of 1994. The study is separated into two stages. In the first stage physical studies of different mechanisms and their reactor dynamic consequences have been analyzed. In the second stage the study is continued with the mitigative actions against inherent boron dilution. This work will include PSA (Probabilistic Safety Assessment, thermal-hydraulic and core-kinetic analyses. Furthermore, the system response and sensitivity to operator actions will be analyzed.

The objects of this paper are to give an overview to inherent boron dilution studies carried out for Loviisa NPP and explain the work to be done in the future.

1. INTRODUCTION

Both the Loviisa PSA study and the deterministic analyses /1/ monstrated that external diluted slugs were potential for leading to severe core damages. During start-up conditions the core can tolerate only limited amounts of pure condensate. This resulted in implementation of temporary process modifications. During the annual refueling outage in 1994 final process modifications against external boron dilution events were implemented.

In Finland boron dilution studies have been extended to cover also inherent boron dilution events. Potential influence of inherent boron dilution events was first discussed by J. Hyvärinen /2,3,4/. During accident and transient conditions boiling margin might be lost and water level may decrease below hot leg level. If secondary side is used for primary loop depressurization or if decay heat is removed to secondary side, primary coolant is condensed in steam generators. This boiling condensing heat-transfer mode is a potential condition leading to diluted slug formation.

In the study of inherent boron dilution the major attention is given to the most potential events leading to core damage. Additionally some pure deterministic scoping studies have been carried out. The two different phases are included in this work:

- thermal-hydraulic and reactor-dynamic studies of initial events leading to inherent boron dilution and core response diluted slugs,
- prevention and mitigation of inherent boron dilution.

The scoping study of inherent boron dilution is divided in three different sub-topics:

- thermal-hydraulic scoping studies leading to inherent boron dilution,
- generic mixing studies,
- core response to diluted slugs.

The first phase of the studies is being carried out to find out initial events and accidents leading to inherent boron dilution and core response to diluted slugs. The main results of this study will be used for reviewing Emergency Operating Procedures. The first phase will be completed during spring 1996 and the reviewing of EOP's are planned to begin during 1996.

2. Thermal-hydraulic scoping studies of inherent boron dilution

The first phase of the study has been concentrated on initial events leading to inherent boron dilution conditions. Although the initial events must be chosen based on PSA, work is mostly dealing with deterministic thermal-hydraulic and reactor-dynamic analyses, so far. The final results will be included in updated PSA-study afterwards.

In the deterministic studies it has been concentrating to find out and identify and understand thermal-hydraulic conditions during inherent boron dilution. Essential phenomena in primary and secondary circuit is presented in figure 1 and table 1.

	PHENOMENA	ESSENTIAL
1	RPV water level	primary circuit natural circulation and boron transportation
2	PRZ line behaviour	water inventory in primary loop
3	void in hot leg	hot leg behavior
4	level in primary collectors	heat transfer mode, natural circulation in horizontal SG's
5	SG water level	heat transfer mode (ATWS,LOFW)
6	void in cold leg	natural circulation, HPI into cold leg
7	PCP coast down	core mass flow rate (ATWS)
8	mixing in downcomer	final inherent safety mechanism against diluted slugs in all accidents
9	break mass flow	convective flows, primary mass inventory, HPI-injection rate
10	core response	burn-up, initial reactivity, reactivity coefficient and boron content are essential in modelling core response
11	role of operator	possible EOP's, role of operators

TABLE 1. Important phenomena in boron dilution transients

The studied initial events were chosen based on both Loviisa PSA study and deterministic thermal-hydraulic studies. Major attention of the thermal-hydraulic analyses is given to:

- SBLOCA (Small Break Loca)
- PRISE (PRImary to SEcondary leakage)
- ATWS (Anticipated Transient Without Scram)
- Initial events leading to severe accidents
- Shut-down conditions

SBLOCA and PRISE are chosen based on Loviisa PSA and previous deterministic studies. Analyses were carried out to find the critical break size leading to boiling-condensing conditions. In 1994, PRISE safety measures were designed to Loviisa NPP. Parallel to that work the inherent boron dilution analyses were carried out to ensure that mitigation against PRISE cases limits also the risk of boron dilution.

During 1994 the Finnish regulatory body,STUK, required IVO to include ATWS analyses in this work regardless to its low probability (10^{-7}-10^{-8}) leading to core failure. Additionally some deterministic studies of severe accident conditions have been included in this work. Extensive scoping studies were carried out with RELAP5-code /5/ parallel with HEXTRAN /6/ calculations (3D neutron kinetics coupled with thermal-hydraulic plant model). Furthermore APROS /7/ thermal-hydraulic and reactor-dynamic system code was used in ATWS scoping studies.

3. ANALYSES OF REACTOR-DYNAMIC RESPONSE TO DILUTED SLUGS

The analysis of core response can be divided in two different parts:

- quasi-stationary analyses
- dynamic analyses

The quasi-stationary analysis was already included in the analyses of external boron dilution analyses and those results were used in these studies too. The object of this study is to find:

- how much diluted water core can tolerate ?
- what is core response to diluted slugs ?

The boundary conditions to core calculation are given by thermal-hydraulic scoping studies. The dynamical core analyses were carried out with HEXTRAN core model coupled with thermal-hydraulic model of both primary and secondary circuits. In SBLOCA analyses the primary circuit thermal-hydraulic boundary conditions was artificially set to correspond to RELAP5 results and the perturbation was described by setting inlet boron concentration.

In spring 1995 extensive ATWS studies were carried out. Regardless of the low probability of ATWS the Finnish regulatory body required IVO to carry out boron dilution studies of ATWS accidents. Although the study started as boron dilution study it was quickly understood that additional studies were needed to explain discrepancies between different code calculations.

The ATWS analyses include two different cases: LOOP (Loss Of Offsite Power) and CRWD (Control Rod Withdrawal). The main object of these studies was to understand fully the thermal-hydraulics and reactor-dynamics and their interactions and the role in boron dilution. It turned out that reactor system will be subjected very quickly, after the coast down of RCP, to boiling condensing heat transfer mode and operators capabilities to mitigate it are rather limited after 10 onset of accidents of accidents. Furthermore there is remarkable risk of missinterprating system behavior.

The studies have continued during summer 1995 to determine what are the capabilities of the operators to cool the reactor safely down if control rods are inserted into the core. In this work both RELAP5 and HEXTRAN code will be used.

Due to the limited understanding of the complicated ATWS mechanisms both point-kinetics, 1D-reactor model and 3D reactor models were used. RELAP5 code and APROS code was used in scoping studies parallel with HEXTRAN-code to understand both thermal-hydraulic and reactor-dynamic feedback mechanisms.

4. NUMERICAL MIXING STUDIES

The first mixing studies using PHOENICS CFD-code (Computational Fluid Dynamic) /8/ was applied to external boron dilution analyses. Experiences from those studies were successfully used in inherent boron dilution mixing studies. However, apparently both additional calculations and some selected experiments will be needed to validate code calculations. Due to the uncertainties related to calculations the mixing studies have been concentrated on to find out the limiting mixing properties.

Parallel with thermal-hydraulic scoping studies some mixing analyses have been carried out. According to the analysis there are apparently mechanisms to generate enough boron diluted water to cause core criticality. On the other hand, it is also known that during almost all accident conditions there exist density differences, heat sinks that may generate density differences leading to buoyancy driven mixing. This seems to be the last and the most effective barrier against inherent boron dilution. Additionally during LOCA situations convective flows may influence the coolant flow so that significant amounts of both steam and boron diluted water may flow out via a break.

The mixing studies are based on both theoretical calculations and thermal-hydraulic mixing experiments. In the first phase the theoretical mixing studies have been concentrating on validating the calculation methods against IVO-PTS (Pressurized Thermal Shock) mixing experiments /9/. In the mixing studies PHOENICS CFD-code has been used. During autumn 1995 and spring 1996 the mixing studies will be continued with parallel mixing code calculations using both PHOENICS and other comercial CFD-codes.

5. REVIEWING EMERGENCY OPERATIONAL PROCEDURES

One of the major tasks of this study was to develop guidelines for the emergency operating procedures. This work has been launched parallel with the thermal-hydraulic scoping studies. The guidelines for updating EOPs will be completed during winter 1996. The major attention is given to the identification of boron dilution conditions and for preventive and mitigative actions in line with present EOP's.

The role of emergency operating procedures is essential. It seems that with correct actions boron dilution can be avoided or, at least, reactor subcriticality can be ensured. Unfortunately it also seems that there still exist possibilities to damage the core by misinterpreting process behavior and/or emergency operating procedures. Emergency operational procedures have not been updated, so far .

The thermal-hydraulic and reactor dynamic studies of inherent boron dilution will be completed in the end of 1995. The main results of, this study will be included in LOVIISA PSA study. Parallel with this, the reviewing of emergency operational procedures will be finalized. The continuation of the studies will be decided after the first phase.

6.Conclusions

In 1993 Finnish Centre for Radiation and Nuclear Safety (STUK) required Imatran Voima Oy to carry out additional boron dilution studies. It was foreseen that boron dilution of external sources is not the only potential mechanism. Furthermore, dilution during transients or accidents can lead to even more dangerous consequences than during normal operation.

The study concentrated on finding out thermal-hydraulig boundary conditions of transients and accidents that leads to inherent boron dilution and its reactor dynamic consequences. The study is based on both PSA and deterministic studies. According to scoping studies and reactor-dynamic studies the boron dilution can not been fully excluded during typical SBLOCA accidents.

During the inherent boron dilution accidents the role of operators are significant. With correct operational actions the reactor can be rather easily to cool down but on the other hand there is possibility to cause severe core damages with misinterpreting the plant behavior and process parameters.

The results of the deterministic scoping studies will be included in Loviisa PSA study parallel with reviewing OAP's.

7.References

/1/ P. Siltanen,M Antila,Extended Protection Against Reactivity Accidents Caused by Slugs of Diluted Water in the Loviisa Reactors,2nd Symposium of AER,PAKS,Hungary,21-26 September,1992

/2/ J. Hyvärinen, Thermal-Hydraulic Safety Impact of Horizontal Steam Generators, International Seminar of Horizontal Steam Generator Modelling, Lappeenranta, Finland, March 11-14th, 1991

/3/ J. Hyvärinen, An Inherent Boron Dilution Mechanism in Presssurized Water Reactors, The Fifth International Topical Meeting on Reactor Thermal Hydraulics, NURETH-5, Salt Lake City,Utah, USA, September 21-24,1992

/4/ J. Hyvärinen, An Inherent Boron Dilution Mechanism in Presssurized Water Reactors, Nuclear Engineering and Design 145 (1993),pp. 227-240

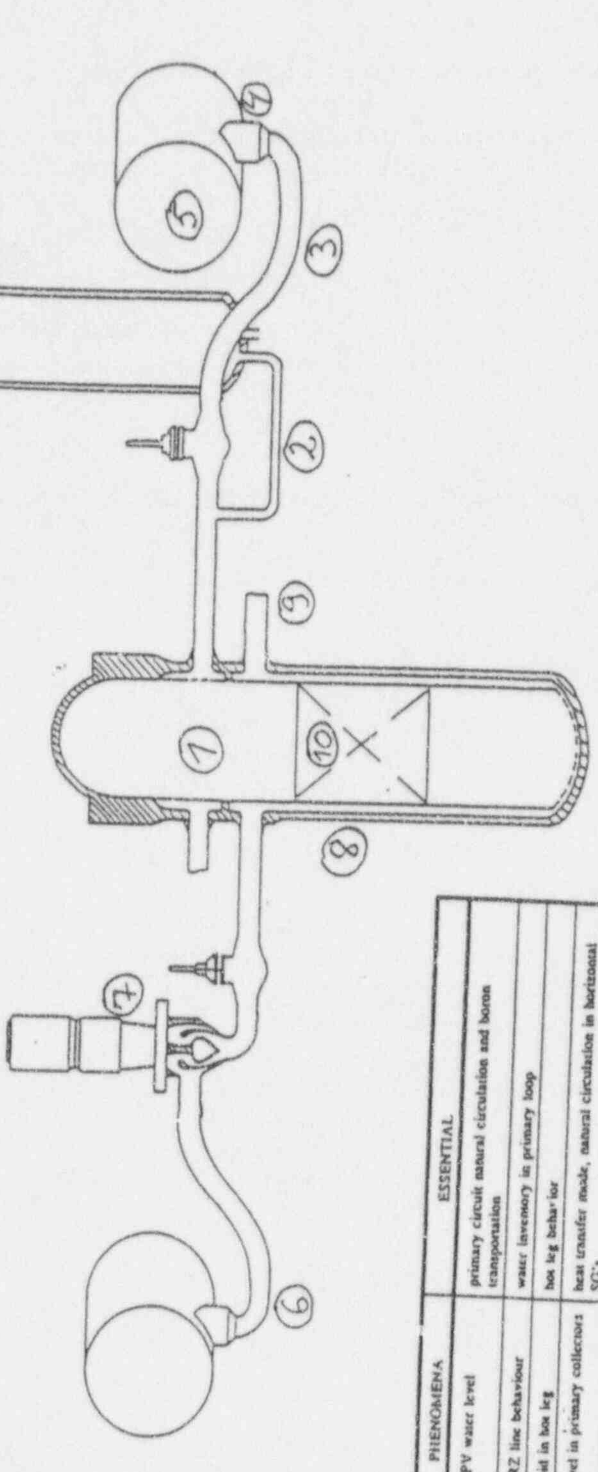
/5/ K.E. Carlson et al, RELAP5/MOD3 Code Manual, Volume 1:Code Structure, System Models and Solution Methods (draft), NUREG/CR-5535,EEG-2596,June 1990

/6/ R. Kyrki-Rajamäki et al,Validation of SMATRA Accident Analysis Code for VVER reactor cores, Proceedings of the International Symposium on Power Plant Transients-1992 ASME Winter Annual Meeting,Anaheim,CA,Nov. 8-13, America Society of Mechanical Engineers,New Yourk,NY,FED-Vol,140,153-162

/7/ K. Porkholm, P. Nurmilaukas, O. Tiihonen, M Hänninen, E.K Puska, Loviisa Nuclear Power Plant Analyzer. Helsinki 1992, Imatran Voima Oy, Research Reports, IVO-A-11/92, ISBN 9512-8928-92-4, 69 pp.

/8/ D.B.Spalding, A Guide to PHOENICS Input Language, CHAM/TR100, 16/07/92

/9/ H.Tuomisto,Experiments and analyses of thermal mixing and stratification during overcooling accidents inpressurizedwater reactor, ANS Proceeding of 24th ASME/AICHE National Heat Transfer Conference, Pittsburgh, Pa, August



	PHENOMENA	ESSENTIAL
1	RPV water level	primary circuit natural circulation and boron transportation
2	PRZ line behaviour	water inventory in primary loop
3	void in hot leg	hot leg behavior
4	level in primary collectors	heat transfer mode, natural circulation in horizontal SG's
5	SG water level	heat transfer mode (ATWS,LOFW)
6	void in cold leg	natural circulation, HPI into cold leg
7	PCP coast down	core mass flow rate (ATWS)
8	mixing in downcomer	final inherent safety mechanism against diluted slugs in all accidents
9	break mass flow	convective flows, primary mass inventory, HPI-injection rate
10	core response	burn-up, initial reactivity, reactivity coefficient and boron content are essential in ,modelling core response
11	role of operator	possible EOP's, role of operators

FIGURE 1.Essential phenomena in inherent boron dilution

BORON MIXING TRANSIENTS IN A PWR. FUNCTIONAL ANALYSIS

J. Feuillet (EDF SEPTEN)

J.J Goetter (EDF SEPTEN)

1 Introduction

EDF has performed different analysis with CEA and FRAMATOME, after Chernobyl accident, to determine criticality accidents which would have not been taken into account for the design.

The conclusion was that a fast reactivity insertion due to the injection of a diluted and / or cold water plug in the core could not be excluded. EDF has considered two severe boron dilution faults which might result in a large reactivity insertion:

- start up of RHR system with unborated water
- start up of a primary coolant pump with a diluted loop.

Probabilistic analysis

Following PRA analysis, it appears that the resulting probability of these events which might lead to a potentially damaging reactivity accident and unacceptable consequences, has been estimated in the range of 10 E-04 10 E-05 per year reactor. The main scenario identified by the PRA analysis is a reactor coolant pump shutdown due to loss of offsite power during reactor coolant dilution by the CVCS.

Physical analysis

If the flowrate in a loop is low enough, a diluted water volume formation is possible in the cold leg, the crossover leg or in the lower plenum. This formation depends on the mixing conditions at the injection point. Conservatively, the analysis carried out by EDF does not account for any mixing at the injection point. Two other important penalizing assumptions are used:

- no mixing of the diluted water in the downcomer is taken into account after a RCP's startup,
- the analysis does not account for any mixing of diluted water during the propagation of the plug in the core.

The maximum diluted water plug volume in the core above which there is a potential risk of fuel damage (prompt criticity reached) was estimated. The maximum volume adopted is 1 m3 whatever the primary boron concentration may be. Taking into account flow dispersion in the downcomer, the maximum volume of diluted water allowable in the cold leg of a static loop was roughly evaluated to 2.5 m3. A new evaluation, currently carried out by EDF LNH with a 3D thermalhydraulic code leads to a value in the range of 3 to 4 m3, provided that neutronic analysis confirms this value.

Conclusions of these first analysis

If we consider the conservative hypothesis recalled above, it is not possible to conclude that the transients described at the beginning of this paper do not induce core damage.

These uncertainties led EDF to decide in 1990 the following measures:

- -Immediately : modifications of operating procedures
- -Short term :
  - -Provisional modifications of automatisms,
  - -Modifications of administrative lock out
- -Mean term :
  - -Elaboration of a research program including thermalhydraulic and neutronic aspects,
  - -Realization of a probabilistic analysis of initiating events in normal operation. Moreover, a functional analysis of initiating events during shutdown maintenance operations has been undertaken,
  - -Proposal of a definitive modification taking into account the conclusions of the probabilistic analysis.

Although the results of the Research and Development program were not available in 1993, EDF decided the realization of definitive modifications and that the corresponding risk should be lower than 10 E-07.

The analysis proposed hereafter focuses on definitive modifications realized on French plants including their consequences on the probabilistic analysis.

2 Description of the definitive modifications

The definitive modification includes:

- -mechanical modifications
- -addition of new automatisms
- -improvement of operating procedures.

2.1 Mechanical modifications

On the 900MW series, the RHRS is kept full of water, when it is isolated from the RCS during normal operation at full power, by a connection with the CVCS letdown. In case of a leakage on the RHRS, RCS water enters the RHRS. The boron concentration of this water, at core end of life, can be very low. The boron concentration difference between RHRS boron concentration and the cold shutdown boron concentration of the RCS can reach 650 ppm.

So it has been decided to modify the filling circuit of the RHRS, which is today connected to the spent fuel pit. The startup of RHRS pump is allowed without chemical and thermal treatment.

On the 1300 MW series, the RHRS has been kept full of water by this mean since the design.

2.2 Addition of automatisms

2.2.1 Dilution function of the CVCS

The dilution function of the CVCS is automatically switched off in case of RCP's shutdown. If RHRS is connected to the RCS, the dilution function of the CVCS is automatically switched off too.

2.2.2 Make up function of the CVCS

The make up function of the CVCS is overseen continuously by an automatism which compares the boron concentrations difference between RCS and CVCS make up line. If this difference reaches 200 ppm, the make up function is switched off if RHRS is connected to RCS.

2.2.3 Boron recycle function

It is automatically switched off in case of RCP's shutdown or if RHRS is connected to the RCS.

2.2.4 Self actuated switch of the suction line of the charging pumps to the Refueling Water Storage Tank

This automatism is actuated after a RCPs shutdown if the thermosiphon is not efficient to mix diluted injected water. The system described on figure 1 does not evaluate today on line the residual energy as a function of the power history of the plant. It is based on correlations between two times T1 and T2 ensuring that if the operation duration at power is greater than T2 or the shutdown duration is lower than T1, the residual heat is sufficient to ensure a good mixing, in case of RCP's shutdown (figure 2). The values T1 and T2 adopted to day on the French 900 MW series are 6 and 35 hours. The most important disadvantage of this modification is that the boration with high borated water delays the rapid startup of the plant, thus degrading Nuclear Power Plant avaibility and finally national grid production capabilities. This last point justifies the research of new methods allowing a more precise evaluation of T1 and T2.

As soon as RHRS is connected, this automatism is disconnected.

2.3 Improvements of operating procedures.

2.3.1 Normal operating procedures

-The setting in service of a demineralizer is not allowed during the filling of the RCS until the time when the first RCP starts up.

-Before the connection of a RHRS loop, the chemical conditionning or a boron concentration measure is required.

-During RCS filling, the charging pumps suction is connected directly to the RWST.

2.3.2 Incidental situations

-Thermal barrier leakage of a reactor coolant pump

The procedures call for closing isolation valves of the thermal barrier fed by the Component Cooling Water System before the primary pressure reaches CCWS pressure.

-Homogeneous dilution procedure is modified

3 Consequences of the modifications described above on the dilution scenarios and on core damage probabilities.

3.1 Initiating events

This chapter describes the possible dilution sources. They can be classified into the following sections:

-Reactivity accident due to diluted and / or cold water addition via CVCS including dilution due to boron removal from the primary coolant;

-Dilution due to the connection of a system or a part of system with a boron concentration lower than the primary coolant one;.

-Dilution due to permeability of a system.

3.2 Dilution via CVCS

A dilution water addition via CVCS is the main risk of reactivity accident occuring in a PWR. The diluted water may be induced by:

-a human error during cooling phases,

-a human error during filling of the RCS,

-loss of off site power and dilution in progress,

-reactivity accident due to cold water addition through the RCP's number 1 seal,

-resin replacement of a demineralizer,

-non effected boron saturation of a mixed beds demineralizer.

3.2.1 A human error during cooling phases.

The RBWMS is used in these situations for a continuous make up of water with the same concentration as the primary one. If the boron concentration is incorrect or if the operator performs a dilution instead of a boration, there will be a dilution incident.

3.2.2 A human error during filling of the RCS

In the new filling configuration required by the procedures, the suction of the charging pumps is connected to the RWST. Before 1990, the suction of the charging pumps during filling of the RCS was connected to the VCT using the automatically make up by the RBWMS.

3.2.3 Loss of off site power and dilution in progress

The provisional modification constitutes a solution against this accident.

3.2.4 Reactivity accident due to cold water addition through the RCP number one seal injection.

The initiating event is a loss of offsite power. The probability of unacceptable consequences is reduced by the switching of the suction line of the charging pumps to RWST if the residual heat is not sufficient to ensure a good mixing. This automatism is redundant.

3.2.5 Resins replacement of a demineralizer (RCS filling in progress)

The initiating event is a resin replacement of a demineralizer cumulated with a human error. (isolation valve of demineralized water not closed). The probability of this scenario is reduced by the improvement of filling procedures.

3.2.6 Non effected boron saturation of a mixed beds demineralizer

Resins in a demineralizer need to be saturated with boron at the primary coolant boron concentration. If this operation is not performed, the diluted water will be injected via the CVCS into the primary coolant system. The probability of this event is reduced by improvements of operating procedures.

3.3 Dilution due to the connection of a system or a part of system with a boron concentration lower than the primary coolant one.

3.3 1 Reactivity insertion via RHRS

The RHR system is set into operation after a conditionning phase. A RCP ensures the primary coolant mixing. When RHRS is out of service during power operation, it may be diluted through heat exchanger leaks or connections with other systems. If the conditionning phase is not performed (incidental transient) a diluted water plug will be injected into the core. The probability of this event has been evaluated to 6 E-O7/ yr. The make up of the RHRS by the spent fuel pit highly reduces this risk.

3.3.2 Reactivity addition via SIS

The initiating event is the discharge of an accumulator after an overpressure test with demineralized water. This event occured in Belleville plant (July 1991). At the present time, procedures call for performing overpressure test with borated water. So the risk due to this sequence should be regarded as very low .

3.3.3 Boron Recycle system

Setting in service of the boron recycle system is no more allowed as soon as RHRS is connected.

3.4 Dilution due to permeability of a system

3.4.1 Leakage through the thermal barrier

The procedure calls for closing isolation valve of the thermal barrier feed on CCS before the primary pressure reaches CCWS pressure.

A leak at low pressure seems to be very unlikely.

3.4.2 CCWS leak through the CVCS heat exchanger

There is a pressure difference between both sides of the seal water return line heat exchanger. A leak from the CCWS to the CVCS might occur. The probability of this scenario is reduced by a modification of the associated procedure.

3.4.3 CCWS leak through RCP's thermal barrier

If the RCS pressure is greater than the CCWS's one, the break of the thermal barrier will lead to a leak from the primary system to the CCW. The thermal barrier will be automatically isolated upon high flow detection. If the RCS pressure is lower than the CCWS's one, there is a leak from the CCWS to the RCS. This scenario will analyzed as a shutdown operation event in the frame of the Reseach and development Program.

3.4.4 Steam Generator tube leak

The formation of a water plug induced by a SG tube leak during cold shutdown for maintenance will also be analyzed in the frame of Research and Development Program.

3.5 Consequences of the previous modifications on core damage frequencies.

A new evaluation of core damage probabilities taking into account definitive modifications has been carried out by EDF. Some result examples obtained in terms of core damage frequency gains are summarized below for each modification.

3.5.1 Automatic control of the boron concentration

- Cooling phases:

Core damage frequency before definitive modification: 10 E-6

Core damage frequency after definitive modification: 10 E-8

-Blow off of the CVCS Volume Control Tank:

Core damage frequency before definitive modification: 10E-07

Core damage frequency after definitive modification: 10E-09

3.5.2 Automatic switching of the CVCS suction pumps from CVCS to RWST

- Cold water plug formation due to RCP's seal injection

Core damage frequency before definitive modification: 7.E-06

Core damage frequency after definitive modification: 2 E-08

3.5.3 Make up of the RHRS from the spent fuel pit

-Reactivity insertion via RHRS

Core damage frequency before definitive modification: 6.E-07

Core damage frequency after definitive modification: << 0

3.5.4 Automatic isolation of the Boron Recycle System after connection of the RHRS to the RCS

-Reactivity insertion via Boron Recycle System

Core damage frequency before definitive modification : 10E-05

Core damage frequency after definitive modification: 10 E-08

3.5.5 Improvements of operating procedures

-Resins replacement of a demineralizer:

Core damage frequency before definitive modification : 2 E-06

Core damage frequency after definitive modification: 2 E-08

- Non borated demineralizer:

Core damage frequency before definitive modification : 5 E-06

Core damage frequency after definitive modification: 1.3 E-07

3.5.6 Conclusion

The analysis of all initiating events shows that the definitive modification reduces the global risk from 2.5E-5 (estimated by EDF with the provisional modification) to 3 E-7 if the definitive modification is considered. This analysis points out the benefits resulting from the improvements of procedures.

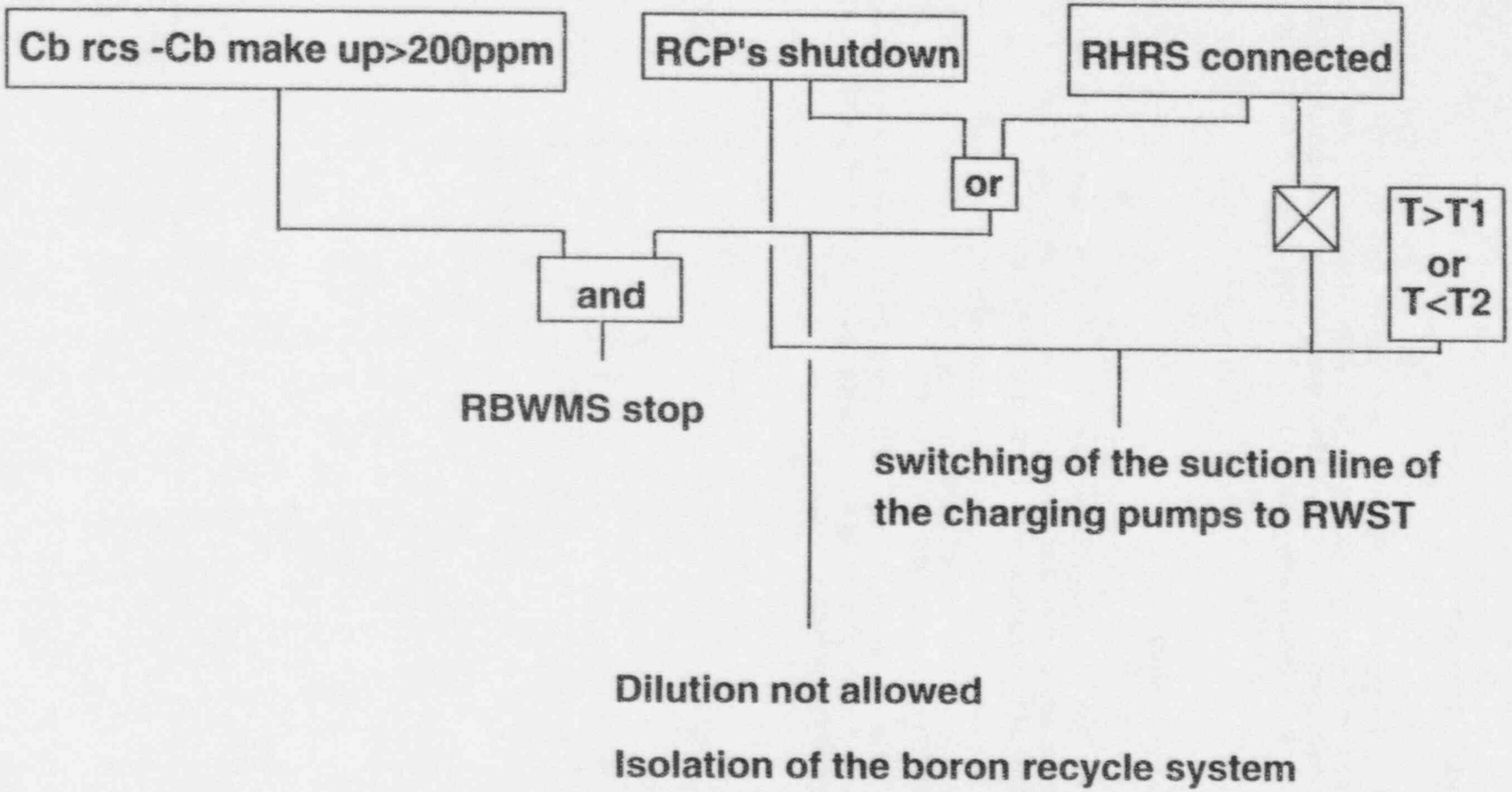
4. Conclusion

The definitive modification realized on the French plants reduces the probability of potential core damage induced by a fast dilution accident. In addition to the palliative measures described above, EDF

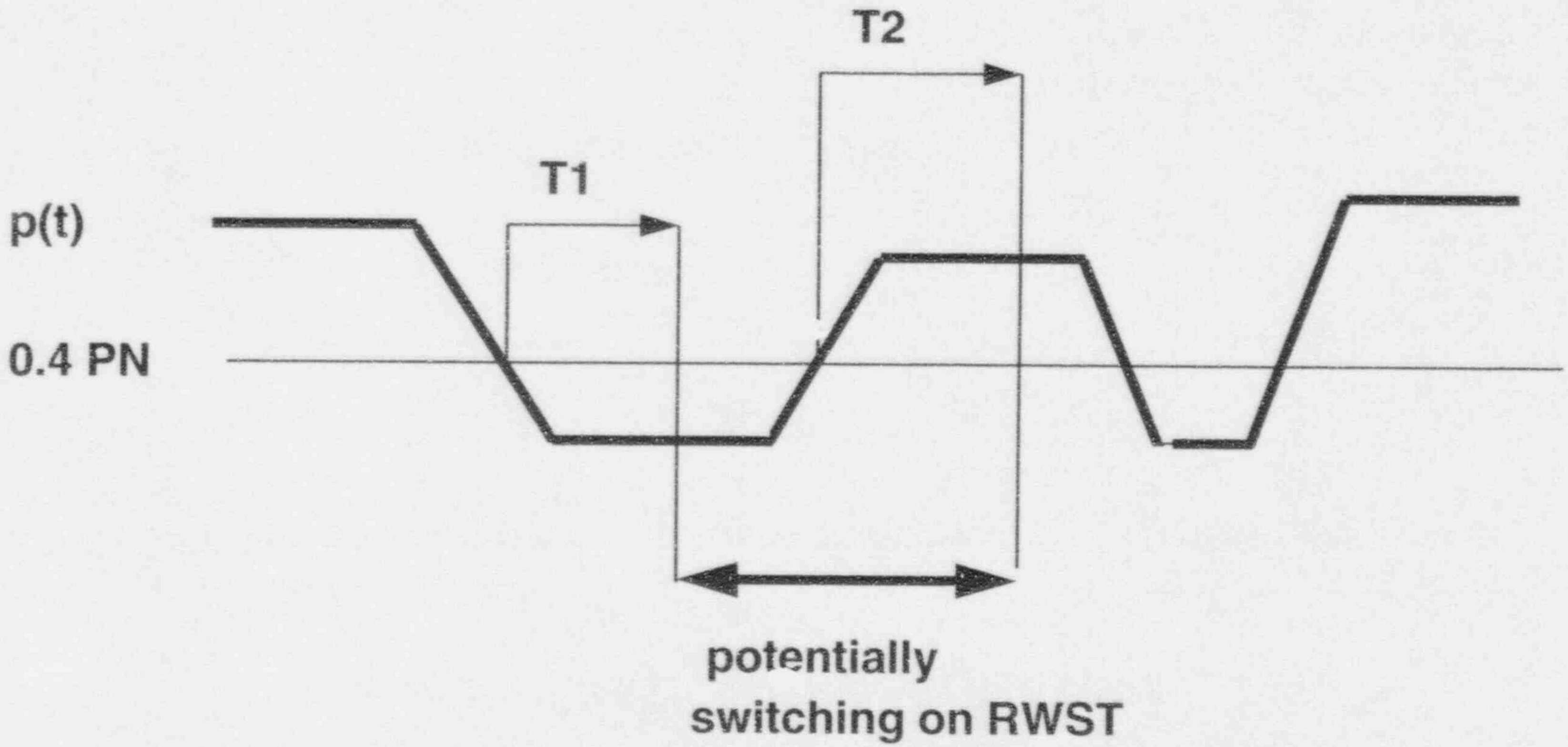
- analyses the dilution phenomena and water plug formation with the aim of acquiring more in-depth understanding of the risk, in particular during cold shutdown for maintenance operations;

- makes a review of the events which might generate the formation of a diluted water plug during maintenance operation and will define, if necessary, associated palliative measures.

MODIFICATION OF AUTOMATISM



TEMPORIZATION T1 AND T2



Probability and Consequences of a Rapid Boron Dilution Sequence in a PWR[1]

D. J. Diamond
P. Kohut
H. Nourbakhsh
K. Valtonen[a]
P. Secker[b]
Brookhaven National Laboratory
Department of Advanced Technology
Upton, NY 11973-5000

Presented at the OECD Specialist Meeting
on Boron Dilution Reactivity Transients
State College, PA
October 18-20, 1995

Background

The reactor restart scenario is one of several beyond-design-basis events in a pressurized water reactor (PWR) which can lead to rapid boron dilution in the core. This in turn can lead to a power excursion and the potential for fuel damage. The scenario occurs during the period when the reactor is being deborated according to normal procedures so that criticality can be achieved. A loss of offsite power (LOOP) is the initiating event. When this occurs, there is reactor trip (the shutdown banks would be withdrawn during deboration) and trip of the charging pumps and the reactor coolant pumps (RCPs). Emergency power is brought on-line quickly, and the charging pumps are energized from the emergency bus. The RCPs remain tripped. It is assumed that the volume control tank (VCT), which supplies the suction for the charging pumps, is filled with a large volume of highly diluted water. This water continues to be pumped into the reactor coolant system (RCS), if the operator takes no action to switch the suction to a borated source. Since the RCPs are not running, if the natural circulation flow rate is low (e.g., soon after a refueling), there is the potential for a slug of diluted water to accumulate locally in the RCS.

The next event in the scenario is the start of an RCP. This only happens after offsite power is restored. It is assumed that the operators will start the RCP in order to resume the restart procedure. When this occurs, it is assumed that the slug passing through the core adds sufficient reactivity to overcome the shutdown margin and cause a power excursion. Furthermore, the concern is that the power excursion is sufficient to cause fuel damage.

---

[1]This work was performed under the auspices of the U.S. Nuclear Regulatory Commission under Contract No. DE-AC02-76CH00016.

[a]Finnish Centre for Radiation and Nuclear Safety, Helsinki, Finland

[b]Department of Nuclear & Energy Engineering, University of Arizona, Tucson, Arizona

A probabilistic analysis had been done for this event for a European PWR. The estimated core damage frequency was found to be high partially because of a high frequency for a LOOP and assumptions regarding operator actions. As a result, a program of analysis and experiment was initiated and corrective actions were taken. A system was installed so that the suction of the charging pumps would switch to the highly borated refueling water storage tank (RWST) when there was a trip of the RCPs. This was felt to reduce the estimated core damage frequency to an acceptable level. In the U.S., this original study prompted the Nuclear Regulatory Commission to issue an information notice[1] to follow work being done in this area and to initiate studies such as the work at BNL reported herein.

In order to see if the core damage frequency might be as high in U.S. plants, a probabilistic assessment of this scenario was done for three plants. Two important conservative assumptions in this analysis were that (1) the mixing of the injectant was insignificant and (2) fuel damage occurs when the slug passes through the core. In order to study the first assumption, analysis was carried out for two of the plants using a mixing model. The second assumption was studied by calculating the neutronic response of the core to a slug of deborated water for one of the plants. All three types of analyses are summarized below. More information is available in the original report.[2]

Probabilistic Analysis

The plants chosen--Oconee, Calvert Cliffs, and Surry--represent a sample from the three U. S. reactor vendors: Babcock & Wilcox, Combustion Engineering, and Westinghouse, respectively. An example of the event trees developed for each of these plants is the one shown in Figure 1 for Calvert Cliffs.

The first top event, ILOOP, is the loss of offsite power initiator and represents the loss of the electrical grid and/or the two 500 kV and the 69 kV transmission lines.

The next top event, DSL, questions the availability of the emergency diesel generators which would provide backup power for the safety systems but not for the RCPs. The diesel generators may fail to start but could recover after a certain period of time, and this is modelled in the top event NR-DSL or nonrecovery probability of the diesel generators. Note that the top branch (i.e., success) under this event represents recovery of the diesel generators.

The charging pump availability is examined at the top event CHG. The recovery of the offsite power is an important event, and P(NR-LOOP) expresses the probability of nonrecovery in a given time interval and is the lower branch (or failure path) on the tree.

The last two top events are related to the condition of the diluted slug and its potential effect on the reactor core. CCD is conditional core damage given that the diluted water has entered the RCS. The RCPRST top event reflects the probability of restarting the RCPs after the LOOP event recovered.

There are three sequences marked in Figure 1 involving core damage potential. The other sequences are unrelated scenarios that do not involve dilution accidents. The three sequences are summarized as follows:

Sequence 1: After a LOOP event, the diesel generators start and the charging flow is automatically reestablished. As soon as offsite power is recovered, the operator restarts the RCPs in a time frame of about 30 minutes. The charging flow is reduced to 44 gpm to extend the time window for LOOP recovery before borating the RCS.

Sequence 2: After the LOOP event, the diesel generators fail to start but recover sooner than the offsite power, and charging flow is automatically restarted. After offsite power recovers, the RCPs may start and core damage may result.

Sequence 3: This is similar to Sequence 2 except the offsite power recovers earlier than the diesel generators. Both charging and the RCPs may be started leading to the dilution event.

These sequences were quantified using data for the initiating event and for the various probabilities that were taken from appropriate sources. Startup after both a nonrefueling and a refueling outage was considered. The difference between the two was primarily the different natural circulation rates expected and the resulting different conditional core damage probability.

The results of the analysis for the three plants are given in Table 1 which shows not only the expected core damage frequency (CDF) but also the initiating frequency. The latter was of particular interest because in the analysis done in Europe, the initiating frequency was quoted as being an order of magnitude higher. The CDF is similar for all three plants and in the range considered significant.

Table 1 Summary of Important Frequencies

	OCONEE		CALVERT CLIFFS		SURRY	
	INIT FR /YR	CDF /YR	INIT FR /YR	CDF /YR	INIT FR /YR	CDF /YR
Refueling	4.93E-5	1.05E-5	6.03E-5	7.54E-6	6.03E-5	3.66E-6
Non-Refueling	1.64E-4	1.75E-5	2.01E-4	1.25E-5	2.01E-4	6.08E-6
Total		2.80E-5		2.00E-5		9.74E-6

These results are dependent on plant design and various assumptions used in the analysis. The most important considerations are summarized below. Note that some of them result in overestimating the core damage frequency.

1. The dilution time during startup is eight hours, but in the analysis, the consequences of the event are independent of when during this period the loss of offsite power occurs. In reality, an event occurring early during this period will have more shutdown margin to overcome and is, therefore, expected to have less of an effect than an event occurring near the end of the normal dilution procedure.

2. No credit is given for the operator to take action and stop the charging flow from the VCT after the LOOP. Although dilution while the shutdown banks are inserted or the RCPs are stopped (as would be the case after a LOOP) is not a normal procedure, it is assumed that since the operator knows that the flow from the primary water makeup pump has ceased, that no other action is deemed warranted. An action that could be taken by the operator would be to switch the charging pump suction to the RWST.

3. For all three plants, the dilution is done with flow from the VCT. It shculd be noted that in some plants the suction for the charging flow comes directly from the primary grade (PG) makeup water source, and once the PG water pump is tripped, there is no longer the potential for adding unborated water to the RCS.

   Oconee: The dilution rate is about the same as the letdown, and the volume in the letdown storage tank is diluted to low boron levels (0-200 ppm). The available volume in the letdown storage tank is about 1900 gal.

   Calvert Cliffs: The dilution rate is matched by the letdown flow rate. The VCT is eventually diluted to a very low boron concentration. The available volume for injection into the RCS is about 2900 gal.

   Surry: The dilution rate is generally lower than the charging rate, and the VCT may get diluted to a very low boron concentration (0-100 ppm). The dilution flow is always directed to the VCT, and bypassing is allowed only during xenon transients. The available volume for injection into the RCS is about 1600 gal.

4. For all three plants, the potential for an accident is limited by the amount of diluted water in the VCT as the supply of primary grade water is stopped by a PG makeup pump. However, there are plants where this pump is connected to the emergency bus, and the probability of an accident will be increased if primary grade water continues to be pumped into the VCT. This appears to be the case for some plants in France and Sweden. At the Ringhals plant in Sweden, of three units designed by Westinghouse, two have the PG water pump connected to the emergency bus. The question of whether the makeup pump trips or continues to run has to be evaluated on a unit by unit basis.

5. Refueling outage: The conditional core damage probability is linearly changing between zero and one, corresponding to the amount of diluted water injected into the RCS. Mixing and switchover to a borated source reduces the probability of core damage from one to zero over a short period of time.

   Non-refueling outage: The probability for conditional core damage varies between zero and one-half to account for the potentially higher natural circulation rate and mixing.

   For any outage, the probability of core damage used is expected to be conservative because it does not account for any mixing that may occur (discussed further below).

6. If offsite power, or another adequate power source, is available, the reactor coolant pumps will be started over a 30-minute interval.

Thermal-Hydraulic Analysis

In the probabilistic analysis, the conservative assumption was made that the charging flow, consisting of highly diluted water, does not mix sufficiently with the borated water in the RCS so that a diluted region accumulates in the lower plenum with the potential to cause a power excursion. It is known that there will be some mixing, and, hence, an attempt was made to quantify the extent of this mixing. The analysis assumes that the unborated charging flow is colder than the water in the RCS, and it is injected into the cold leg which is otherwise stagnant or at a low natural circulation flow rate.

The modelling approach is similar to that used in the regional mixing model developed by Nourbakhsh and Theofanous[3,4,5]. That work was in support of the NRC Pressurized Thermal Shock (PTS) study to predict the overcooling transients due to high pressure safety injection into a stagnant loop of a PWR. The analysis includes quantification of mixing (entrainment) at locations where the mixing is expected to be intense, such as at the connection of the charging line to the reactor coolant system and in the downcomer. These mixing models are then used to determine the dilution boundary as a function of time.

Qualitatively, the physical condition may be described with the help of Figure 2. In the absence of loop flow, the relevant parts of the system include the loop seal, pump, cold leg, downcomer, and the lower plenum. Initially, this portion of the primary system is filled with borated water with a boron concentration of ~1500 ppm and at a temperature near that of normal operation (~290°C or ~550°F). The dilution transient occurs with charging pump(s) injecting unborated water into the cold leg at a rate of ~45 to 96 gpm. Typical temperatures of the makeup (charging) flow are 200°C to 260°C (400°F to 500°F) although, depending on the plant and the stoppage of letdown flow during LOOP, lower temperatures, on the order of 70°C (160°F) are also possible.

The ensuing flow regime is schematically illustrated in Figure 2. A "cold diluted stream" originates with the charging buoyant jet at the point of injection, continues toward both ends of the cold leg, and decays away as the resulting buoyant jets fall into the downcomer and pump/loop-seal regions. A "hot stream" flows counter to this "cold diluted stream" supplying the flow necessary for mixing (entrainment) at each location. This mixing is most intensive in certain locations identified as mixing regions (MRs). MR1 indicates the mixing associated with the highly buoyant charging jet. MR3 and MR5 are regions where mixing occurs because of the transitions (jumps) from horizontal layers into falling jets. MR4 is the region where the downcomer (planar) buoyant jet finally decays. The cold streams have special significance since they induce a global recirculating flow pattern with flow rates significantly higher than the charging flow. The whole process may be viewed as the quasi-static decay of the cold diluted stream within a slowly varying "ambient" temperature and boron concentration.

The Regional Mixing Model, which has been developed to study the thermal mixing of interest to pressurized thermal shock, was utilized to assess the extent of boron mixing in the absence of loop flow during a reactor restart scenario. Illustrative reactor predictions for Surry and Calvert Cliffs indicate significant mixing during the boron dilution transients.

Indeed, for the cases considered, the boron concentration in the lower plenum does not fall below 900 ppm. However, these cases do not encompass all possible physical situations for these plants. It would also be desirable to assess the applicability of the model when the temperature of the charging flow is higher, to improve the understanding of mixing when the injectant enters at the side or bottom of the cold leg piping, and to study mixing under low natural circulation flow.

Analysis of Consequences

A synthesis method was used to study the core response. Static 3-dimensional core calculations are combined with point neutron kinetics calculations to determine the power excursion. The static calculations determine the slug reactivity which is input to the power calculation. In lieu of doing detailed boron transport calculations, different slug geometries and boron concentrations are assumed. The slug is assumed to enter the core uniformly across either the entire inlet area or across only a section of the inlet. The slug reactivity is calculated as a function of the position of the slug front, and a constant speed is assumed in order to translate the space dependence into a time dependence.

The neutron kinetics model is combined with a heat conduction model in order to improve the accuracy of the fuel temperature calculation relative to an adiabatic model. This model also calculates the core average fuel enthalpy. At the time at which the fuel enthalpy is at a maximum, the position of the slug front is noted, and the corresponding static calculation is used to determine the power peaking factor. The local peak fuel enthalpy is then calculated by adding to the initial core average fuel enthalpy the increase in core average fuel enthalpy multiplied by the power peaking factor. The local peak fuel enthalpy can then be compared to the criterion for catastrophic fuel damage. This is taken to be 280 cal/g (1.2 MJ/kg) which is the acceptance criterion for design-basis reactivity-initiated accidents calculated for licensing. The enthalpy criterion may change in the future because of concerns about high burnup fuel behavior. In that case, these results could be reinterpreted with the new criterion.

Note that if there is no catastrophic fuel damage, there is still the possibility of release of fission products due to cladding damage either because of stresses caused at lower enthalpies or because of dryout on the surface of the clad. There is also the possibility of a pressure increase that could be excessive under shutdown conditions.

Boron dilution reactivity as a function of time was taken from static calculations for a model representing Calvert Cliffs. The constant speed used for the slug front was 61 cm/s (2.0 ft/s) which corresponds to 13 percent of rated flow. This is an approximation to the flow which would increase from close to zero (assuming little natural circulation) to 20 percent of full flow in about 20 seconds.

A constant negative reactivity representing the initial shutdown margin was used. This is meant to account for the worth of the shutdown banks in the reactor startup scenario and, in general, for any other contribution to shutdown margin that might be present before the dilution begins. In the present calculation, 4 percent shutdown is assumed to be the base initial condition. In Calvert Cliffs, this is approximately equal to the shutdown bank worth, but in other plants, the shutdown worth might be smaller.

Fuel temperature reactivity was calculated during the dynamic simulation using the core average fuel temperature calculated by the model and a Doppler feedback coefficient expressed per unit change in square root of absolute temperature. The Doppler feedback is strongest in the region where the fuel temperature is highest and since the power, and hence the neutron importance, is also highest in this region, it is a gross inaccuracy to represent the Doppler feedback using a core-average fuel temperature. To improve upon the accuracy of the fuel temperature feedback, a Doppler weighting factor (DWF), obtained from static calculations, was applied to the reactivity calculated using the core average fuel temperature.

The power response for a typical event is shown in Figure 3. It consists of a sharp (prompt-critical) power rise after the slug has moved into the core. This is terminated by Doppler feedback. The power then rises again due to the fact that the slug is still passing through the core, and approximately two seconds later the power decreases as the slug leaves the core. There is also a decrease in fuel enthalpy primarily from the fact that there is heat transfer from the fuel into the coolant, and this becomes appreciable in the period after 6 seconds. Note that when there is a large amount of energy transferred into the coolant, the model is no longer applicable. Two-phase flow would be expected, and significant negative feedback from the coolant would affect the response of the fuel rods. Although this was not modeled in this calculation, the peak fuel enthalpy was calculated during the period of 5-6 seconds when the core average fuel enthalpy reaches a peak.

These results will be most sensitive to the initial shutdown margin, the Doppler feedback, and the properties of the slug. Other factors which will have a secondary impact are the delayed neutron fraction, the neutron lifetime, and the speed at which the slug moves through the core.

The initial shutdown margin represents what the condition of the core might be by virtue of the operating mode combined with the effect of any control rods that might insert prior to the slug passing through the core. In the reactor restart scenario, even if the core was initially at its final boron concentration, there would still be some negative reactivity as the reactor is usually brought to critical on the movement of the regulating banks. Assuming that the reactivity hold-down of the regulating banks is small, then the initial shutdown margin is just the worth of the shutdown banks which would scram when there was a loss of offsite power. Although this was assumed to be 4 percent in the base case, for some plants this might only be 2 percent. If there were no rods initially withdrawn, then the smallest shutdown margin that could be expected would be the 1 percent requirement when at hot shutdown conditions. Hence, the calculations were done with an initial shutdown margin down to 1 percent. At the other end of the scale is the fact that if the shutdown margin were equal to or greater than the amount of reactivity which could be inserted by the diluted slug, then no power excursion can take place. In the case being considered with a 750 ppm slug, this is 5.9 percent.

The results of these calculations are shown in Figure 4. Peak fuel enthalpy is plotted for the time corresponding to immediately after the initial power spike and for the time at which the core average enthalpy exhibits a broad maximum. The times at which these peaks occur become later with an increase in initial shutdown margin as it takes longer to overcome that barrier and become supercritical. As can be seen, when the initial shutdown margin is between 1 percent and 2 percent, the peak fuel enthalpy can exceed the 280 cal/g criterion for catastrophic fuel damage. This is not the result of the initial power burst but rather the fact that the power remains high due to the continued presence of the diluted slug.

Figure 4 also shows the effect of reducing the Doppler feedback by a factor of 0.5. The reduction in the Doppler coefficient from -1.4 pcm/°C to -0.7 pcm/°C is consistent with the range of coefficients expected during operation of this cycle of Calvert Cliffs and similar to how other PWRs operate.

An estimate of the boron dilution necessary to cause the fuel enthalpy to exceed 280 cal/g when the initial shutdown margin is 4 percent can be made by using the results shown in Figure 4. Since that figure shows that a reduction of the shutdown margin to approximately 1.5 percent would cause the peak fuel enthalpy to exceed 280 cal/g, it can be estimated that at an initial shutdown margin of 4 percent one would need additional reactivity worth 2.5 percent, or approximately an additional dilution of 320 ppm. Hence, if the boron concentration of the slug was approximately 430 ppm and the initial shutdown margin was 4 percent, catastrophic fuel damage might be likely. Note that with this boron concentration, the volume of the diluted slug would be 1.4 times the volume of the unborated water assumed to be available from the VCT or 10.6 m^3 (375 ft^3). The length of the slug would then be approximately 2.1 m (7 ft) rather than the 3.0 m (10 ft) for the nominal case. This is not expected to alter the behavior of the power excursion during the period w[illegible]e fuel enthalpy reaches a maximum; however, if the slug was even smaller and m[illegible] it is not clear that the situation would lead to a higher fuel enthalpy. The compe[illegible]een a higher positive reactivity insertion and a shorter insertion time in the limit of [illegible]ume leads to a smaller effect.

Summary

The analysis of a rapid boron dilution event has been carried out in three different ways: (1) a probabilistic analysis for the core damage frequency, (2) a mixing analysis to determine the extent of dilution before injection into the core, and (3) neutronics calculations to determine core behavior and the consequences in terms of fuel damage.

The probabilistic analysis was done for reactors from each of the three U.S. reactor vendors. The CDF varies from 9.7 to 2.8E-5/yr for the three plants which is what is calculated for other internal events that are considered important. The analysis shows the importance of the primary grade water pump. For all three plants, the potential for an accident is limited by the amount of diluted water in the VCT as the supply of primary grade water is stopped by the trip of the PG makeup pump. However, there are plants where this pump is connected to the emergency bus, and the probability of an accident will be increased if primary grade water continues to be pumped into the VCT.

For all three plants, the dilution is done with flow from the VCT. In some plants, the suction for the charging flow comes directly from the primary grade makeup water source, and once the PG water pump is tripped, there is no longer the potential for adding unborated water to the RCS.

The results are dependent on assumptions used in the analysis three of which are summarized below. These assumptions result in overestimating the core damage frequency.

1. The dilution time during startup is 8 hrs. The consequences of the event are assumed independent of when during this period the loss of offsite power occurs. In reality, an event occurring early during this period will have more shutdown margin to overcome and is, therefore, expected to have less of an effect than event occurring near the end of the normal dilution procedure.

2. No credit is given for the operator to take action and stop the charging flow from the VCT after the LOOP. Although dilution while the shutdown banks are inserted or the RCPs are stopped (as would be the case after a LOOP) is not a normal procedure, it is assumed that since the operator knows that the flow from the primary water makeup pump has ceased, that no other action is deemed warranted. An action that could be taken by the operator would be to switch the charging pump suction to the RWST.

3. The analysis does not account for any mixing at the point of injection or mixing of the diluted water due to its flow into the downcomer and down to the lower plenum. It does, however, approximately account for the effect of natural circulation on mixing. Mixing analysis performed for this study shows that this may be very conservative under certain conditions.

Mixing analysis was done with the Regional Mixing Model developed to study the thermal mixing of interest to pressurized thermal shock. Illustrative predictions for reactor designs from two vendors show significant mixing during the event. For the cases considered, the boron concentration in the lower plenum does not fall below 900 ppm when the initial boron concentration in the vessel is 1500 ppm. However, these cases do not encompass all possible physical situations for these plants. It would be desirable to assess the applicability of the model when the temperature of the charging flow is higher, to improve the understanding of mixing when the injectant enters at the side or bottom of the cold leg piping, and to study mixing under low natural circulation flow.

The neutronic results show that there is the possibility of catastophic fuel damage depending on (1) the initial shutdown margin, (2) the Doppler feedback, and (3) the properties of the slug, especially the boron concentration. The intial shutdown margin depends on the reactor state at the time of initiation of the event and the reactivity worth of the shutdown bank which will scram before the slug enters the core. The Doppler feedback is responsible for initially terminating the power excursion. This number can vary by a factor of 2 during a fuel cycle, and ,therefore, results will be sensitive to where in the fuel cycle the event takes place. After the initial power excursion, the power remains high until the slug has passed sufficiently through the core so that power decreases. Since there may not be much shutdown margin to begin with, after the slug passes through the core, the decline in power may not be as rapid as occurs with reactor trip. This also impacts the consequences in terms of fuel damage.

References

1. "Foreign Experience Regarding Boron Dilution," Information Notice 91-54, U.S. Nuclear Regulatory Commission, September 6, 1991.

2. D.J. Diamond et al., "Probability and Consequences of Rapid Boron Dilution in a PWR," NUREG/CR-5819, Brookhaven National Laboratory, June 1992.

3. T. G. Theofanous and H. P. Nourbakhsh, "PWR Downcomer Fluid Temperature Transients Due to High Pressure Injection at Stagnated Loop Flow," *Proc. Joint NRC/ANS Meeting, Basic Thermal Hydraulic Mechanisms in LWR Analysis*, September 14-15, 1982, Bethesda, Maryland, NUREG/CR-0043, p. 583, U. S. Nuclear Regulatory Commission, April 1983.

4. H. P. Nourbakhsh and T. G. Theofanous, "Decay of Buoyancy-Driven Stratified Layers with Applications to Pressurized Thermal Shock, Part I: The Regional Mixing Model," *Nucl. Eng. & Design*, in press, 1995.

5. T.G. Theofanous, H.P. Nourbakhsh, P. Gherson, and K. Iyer, "Decay of Buoyancy-Driven Stratified Layers with Applications to Pressurized Thermal Shock," NUREG/CR-3700, U.S. Nuclear Regulatory Commission, May 1984.

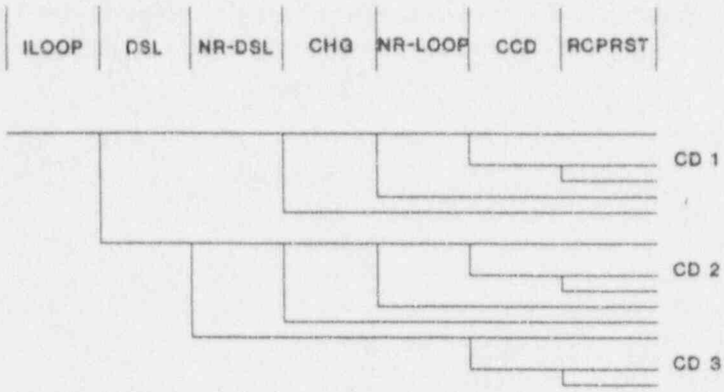


Figure 1 Boron Dilution Event Tree - Calvert Cliffs

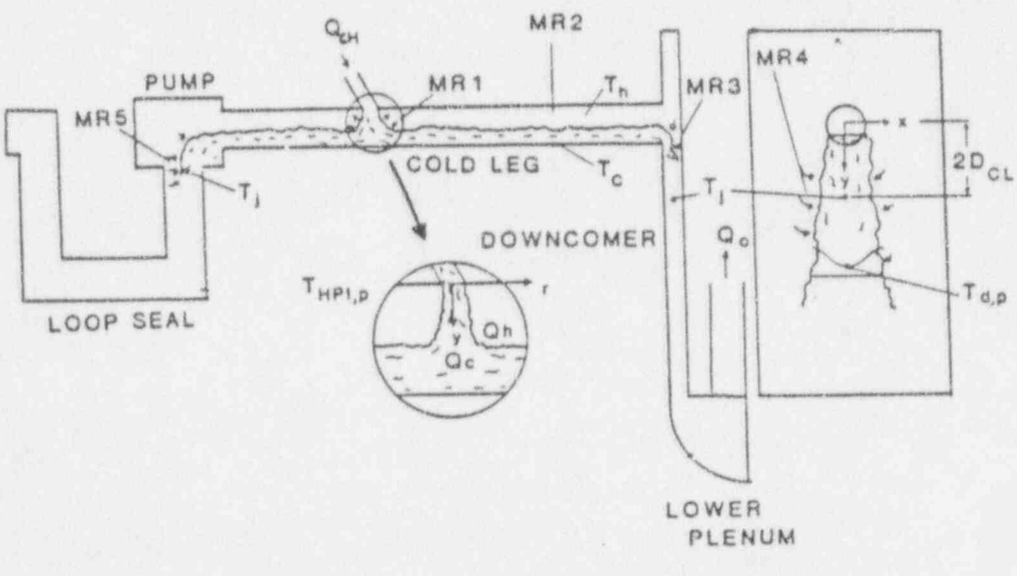


Figure 2 Schematic of the Flow Regime and Regional Mixing Model

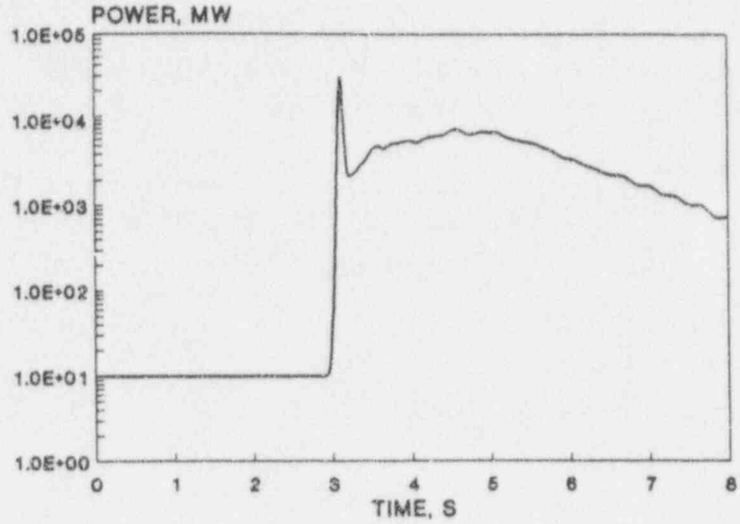


Figure 3 Core Power with Finite Slug

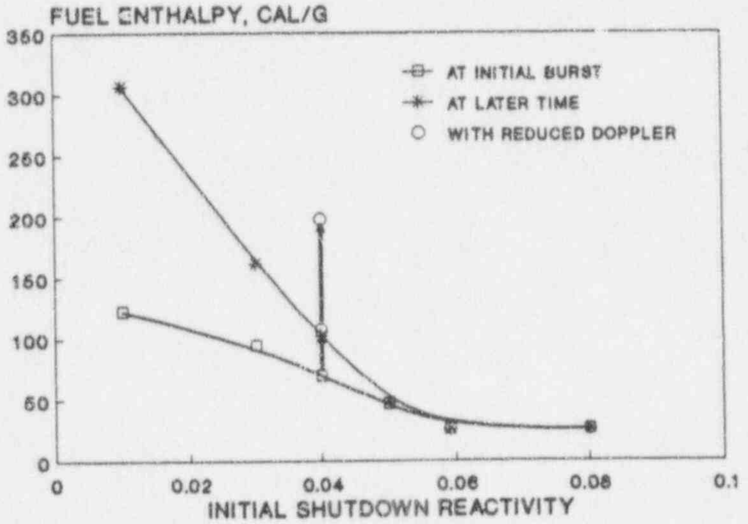


Figure 4 Peak Fuel Enthalpy vs. Shutdown Margin

OECD/CSNI Specialists Meeting on Boron Dilution Reactivity Transients
State College, PA, USA, October 18-20, 1995

"Determination of Reactivity Insertion Effects due to Local Boron Dilution"[1]

M. Clemente, S. Langenbuch
Gesellschaft für Anlagen- und Reaktorsicherheit (GRS) mbH
85748 Garching, Germany

Abstract

In safety studies for shutdown conditions of NPPs, various event sequences were identified which could lead to a local boron dilution by a plug of pure water or of reduced boron content transported to the reactor core. The consequences of such a reactivity initiated accident are mainly determined by the amount of positive reactivity insertion. Therefore, calculations have been performed using a 3D reactor core model to determine the reactivity insertion in shutdown conditions for various assumptions on the volume of the plug and the degree of mixing. The results quantify the minimum volume of a plug and its dependence on subcriticality and mixing conditions which could lead to fuel failures. Thus, the results can be useful for evaluating various boron dilution scenarios.

1 Introduction

Safety studies for low power and shutdown conditions of NPPs performed in France, the USA and also in Scandinavian countries /1, 2, 3, 4/ directed the interest to accident sequences for which a local boron dilution in the reactor core by a plug of pure water or of reduced boron content might occur. The evaluation of such events needs a thorough review of initiating events for all possible operational states and a detailed analysis of system design features including operational procedures. The investigations show that especially during shutdown conditions the plant states have to be differentiated with respect to the different phases from subcritical hot zero power to cold

[1] Work performed under contracts with Federal Ministry for Environment, Protection of Nature and Reactor Safety.

C:\AMIPRO\DOCS\LAB P-OECD95.SAM

conditions for fuel assembly reload. These system analyses are the basis for defining specific accident sequences including postulated malfunctions and their frequency of occurrence. Complementary to the present preliminary state of these investigations, it is of interest to estimate the consequences of postulated local boron dilution accidents by analyzing the reactivity behaviour of the reactor core. The main objective of the calculations performed was to determine the reactivity insertion effect caused by a plug of water under various assumptions on the distribution within the core and the degree of mixing before reaching the core inlet.

2 Scenarios of boron dilution events

The investigations of low power and shutdown conditions mainly consider two groups of events /5, 6, 7, 8/:

- Events with consequences for the residual heat removal by a reduced inventory of primary coolant or by a total loss of heat removal capacity,
- Events with consequences for the subcriticality condition by reducing the effective boron concentration.

Typical events for the second group with reduction of boron concentration are, e.g.,

- Injection and accumulation of demineralized water in a part of the primary circuit and the transport of this volume into the core by the start-up of the corresponding coolant pump. This scenario is discussed occurring during the start-up making the core critical in case of a temporary loss of power supply.
- Cooldown of the plant under emergency power conditions with an isolated steam generator - the corresponding loop flow stagnation could prevent the increase of boron concentration which is necessary for cooldown - or with a leak flow from the secondary to the primary side due to a tube rupture in the steam generator.
- The reflux condenser mode of operation during a small break loss-of-coolant accident.
- Dilutions in trains or components of ECCS or residual heat removal systems.
- Dilutions during reload activities.

C:\AMIPRO\DOCS\LAB P-OECD95.SAM

Typically, the accident scenarios of boron dilution consist of several phases. The first step is a failure or malfunction that generates a volume of pure water or a volume with reduced boron content which may be injected into the primary circuit or transported to the reactor core. The transport to the reactor core occurs by starting a main coolant pump or by initiating natural circulation conditions after a phase of flow stagnation. Along the way flowing to the reactor core the volume may be diminished or partly mixed in the primary circuit, especially flowing through the downcomer and the lower plenum. Finally, a plug of demineralized water or a volume of reduced boron concentration will reach the reactor core and cause a positive reactivity insertion.

3 Reactivity effects due to local boron dilution

The amount of reactivity insertion due to the boron dilution determines the effect on core behaviour. If the reactivity change is low, subcriticality will be maintained by the inserted control rods. If the positive reactivity insertion is sufficiently high, recriticality could occur. However, severe consequences would be possible only if the excess reactivity reaches prompt critical values.

The reactivity effects in the reactor core by a postulated boron dilution are calculated by a 3D reactor core model. The analysis is performed by using the reactor core model QUABOX/CUBBOX /9/, which solves the neutron diffusion equations with two energy groups in 2D- or 3D-geometry by a coarse mesh method based on local polynomial approximation of the neutron flux. The basis of investigations is a typical core loading of a PWR plant. The initial state is a critical core configuration at hot zero power. Then, for the shutdown condition, with all control rods inserted, a disturbance of the initially homogeneous boron concentration is assumed and the corresponding reactivity change is calculated. Two different types of spatial distribution are considered:

- A symmetric distribution assuming that the change of boron concentration is homogeneous across the core area where the volume is varied by adjusting the axial height.
- An asymmetric distribution assuming that the change of boron concentration occurs in a local region at the core periphery where the volume is varied by increasing the radial area.

C:\AMIPRO\DOCS\LAB P-OECD95.SAM

A series of calculations are performed for both assumptions by varying the boron concentration in the disturbed region. The limiting condition is defined by assuming pure demineralized water. If a certain degree of mixing occurs before the volume enters into the core region, then the boron concentration reaches a fraction of the initial boron concentration in the primary circuit.

The results for a symmetric distribution in hot zero power condition are shown in fig. 1. The critical boron concentration in hot zero power is about 1080 ppm. The figure presents the reactivity insertion versus plug volume. Reactivity values have been calculated for a pure plug corresponding to 100% boron dilution and for conditions assuming a certain degree of mixing corresponding to 90% or 80% boron dilution, respectively. Two lines are included: The first line indicates the shutdown reactivity in hot zero power condition. The second line indicates the reactivity which was estimated as limit value causing fuel rod failures. This estimation is based on transient calculations using a pointkinetics model and a failure criteria of maximum fuel enthalpy of 200 cal/g.

The assumptions for the asymmetric distribution in the radial plane are described in the core geometry, fig. 2, where two different volumes at the core periphery are marked. The calculated reactivity values for an asymmetric distribution in hot zero power conditions are shown in fig. 3. The same notation as in fig. 1 applies.

A comparison of results indicates that a plug of pure demineralized water in the asymmetric condition is more efficient in reactivity change than in the symmetric condition. Recriticality in hot zero power may occur by a plug volume of about 0.5 m^3 for asymmetric conditions and by a volume of about 3 m^3 in the symmetric condition. The limit reactivity value of fuel failures may be reached in the asymmetric condition by about 1.5 m^3 and in the symmetric condition by about 5.0 m^3. Both dependences show that the highest contribution to the reactivity change is already caused by small volumes. The reactivity insertion reaches asymptotic conditions for larger volumes. The reactivity change is reduced assuming a certain degree of mixing. If boron dilution is limited to 80% in hot zero power, then a recriticality would occur, but the estimated reactivity value corresponding to fuel rod failures would not be exceeded.

Also for cold conditions, when the boron concentration is increased to 2200 ppm, the reactivity change by a boron dilution has been calculated. The results for this condition are presented in fig. 4.

C:\AMIPRO\DOCS\LAB P-OECD95.SAM

4 Conclusions

It is a generally implemented nuclear design feature to limit values of positive reactivity insertion to exclude fast reactivity initiated accidents. For a local boron dilution accident, there exists the potential of a high reactivity insertion under unfavourable conditions. The reactivity values have been estimated for various assumptions on the plug volume and the degree of mixing. If a certain degree of mixing can be assured, then the consequences would be acceptable. But, as the consequences of a pure plug could be severe, even in shutdown conditions, such conditions by external injection of demineralized water have to be avoided by system design. When the core is loaded, all plant states in shutdown condition have to be considered including plant states during maintenance. Work is going on to develop more realistic models by applying multidimensional flow models for the vessel and by coupling 3D neutronics models with system codes for analyzing integral plant behaviour. These modelling efforts should be supported by experimental investigations to study relevant mixing effects. In addition, detailed system analyses should be continued to determine plant specific accident conditions.

5 References

/1/ Champ M.
Risk Evaluation of Dilution Accidents
Rapport S.A. C. P. No 122, Dec. 1990

/2/ Diamond D J., Kohut P., Nourbakhsh H., Valtonen K., Secker P.
Probability and Consequence of Rapid Boron Dilution in a PWR
NUREG/CR-5819, BNL-NUREG-52313, June 1992

/3/ Jacobsen S.
Effects at Local Dilution Transients
Sweden, Ringhals Nuclear Power Plant, 1989

/4/ Hyvärinen J.
An Inherent Boron Dilution Mechanism in Pressurized Water Reactors
CSNI Specialist Meeting on Transient Two-Phase Flow, Current Issues in System Thermal-Hydraulics, 06.-08.04.1992, Aix-en-Provence

/5/ IAEA
Technical Committee Meeting on 'Modelling of Accident Sequences during Shutdown and Low Power Conditions'
Stockholm. 30.11-03.12.1992

/6/ IAEA
Technical Committee Meeting on 'Procedures for Probabilistic Safety Assessment for Shutdown and Other Low Power Operating Modes'
Arnheim, The Netherlands, 8.-11.11.1994

/7/ Simon M.
Spezielle Risiken und Unfallszenarien im Anlagenstillstand, Sicherheit von Kernkraftwerken im Stillstand
SVA-Kurs, 29.-31.03.1995, Brugg-Windisch

/8/ Simon M.
Scenarios of Boron Dilution Incidents and Concerns regarding Plant Safety
Specialists Meeting on Boron Dilution Reactivity Transients

/9/ Langenbuch S., Maurer W., Werner W.
Coarse-Mesh Flux-Expansion Method for the Analysis of Space-Time Effects in Large Light Water Reactor Cores
Nuclear Science and Engineering, 63, pp. 437-456, 1977
High-Order Schemes for Neutron Kinetics Calculation Based on a Local Polynomial Approximation
Nuclear Science and Engineering 63, pp 508-516, 1977

C:\AMIPRO\DOCS\LAB P-OECD95.SAM

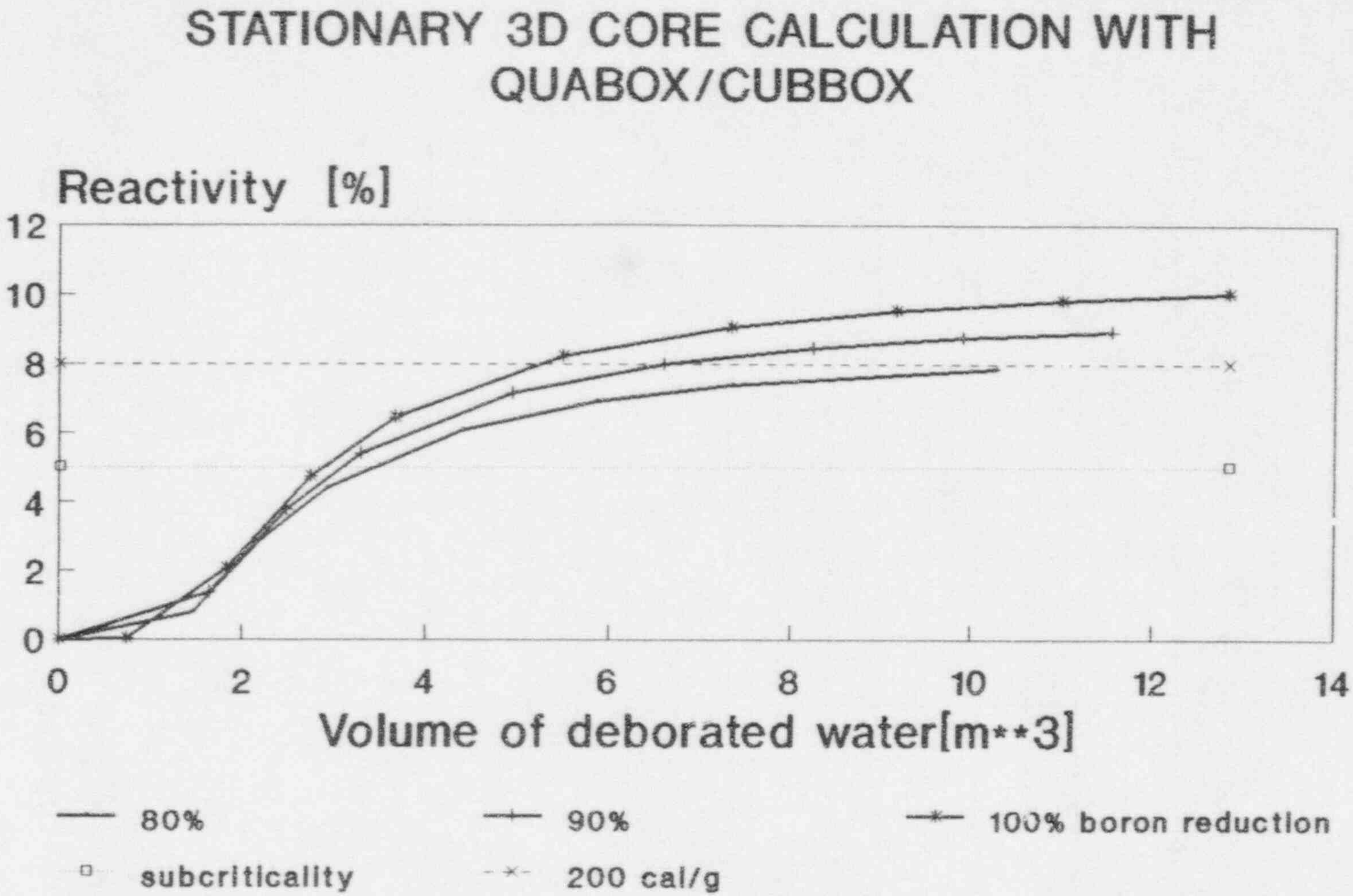


Fig. 1 Symmetrical boron dilution
hot zero power
critical boron concentration: 1082ppm

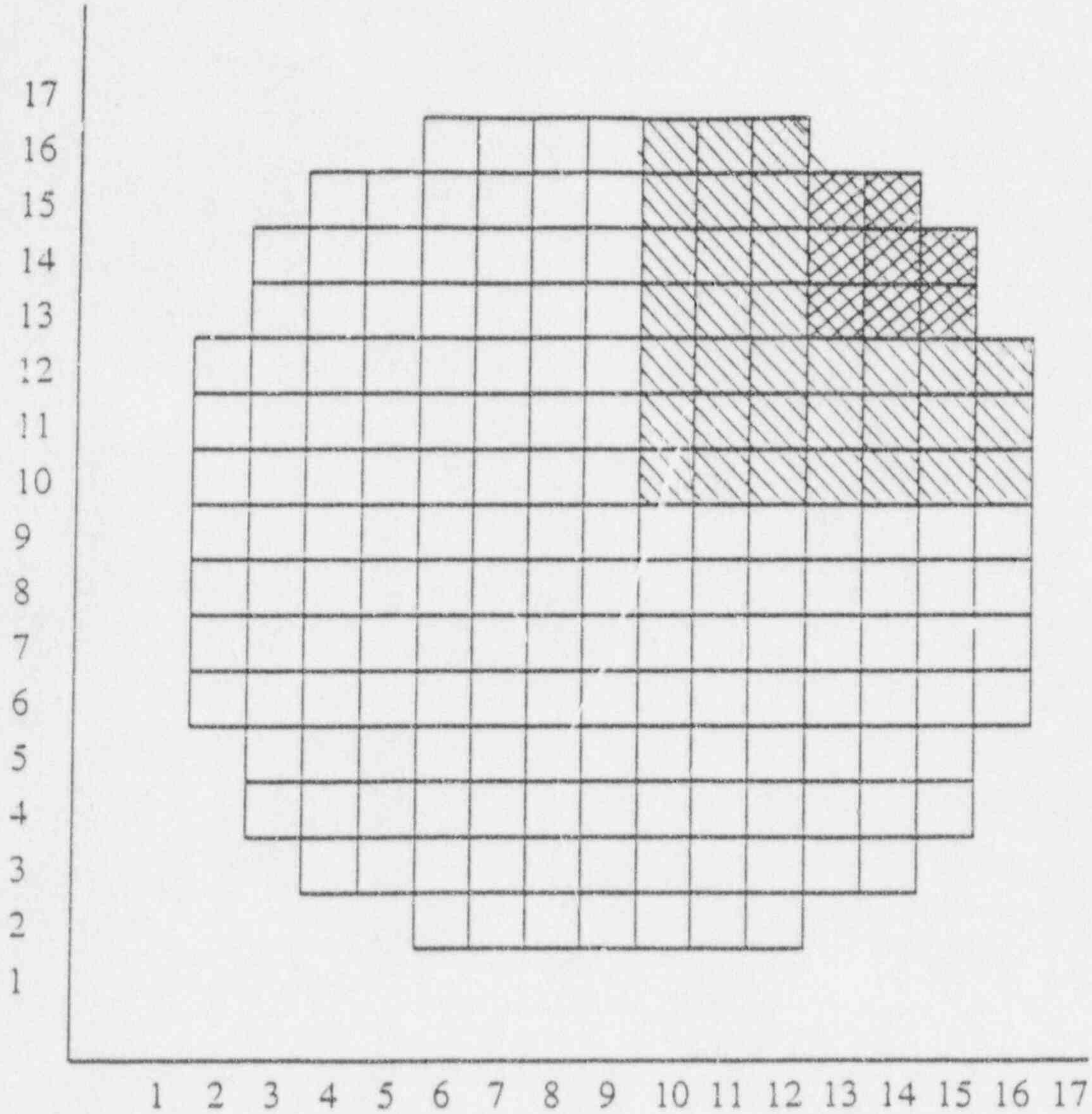


Fig. 2

Definition of core regions with reduced boron concentration. The marked regions correspond to free coolant volumes of 4.67 and 0.91m^3 respectively.

STATIONARY 2D CORE CALCULATION WITH QUABOX/CUBBOX

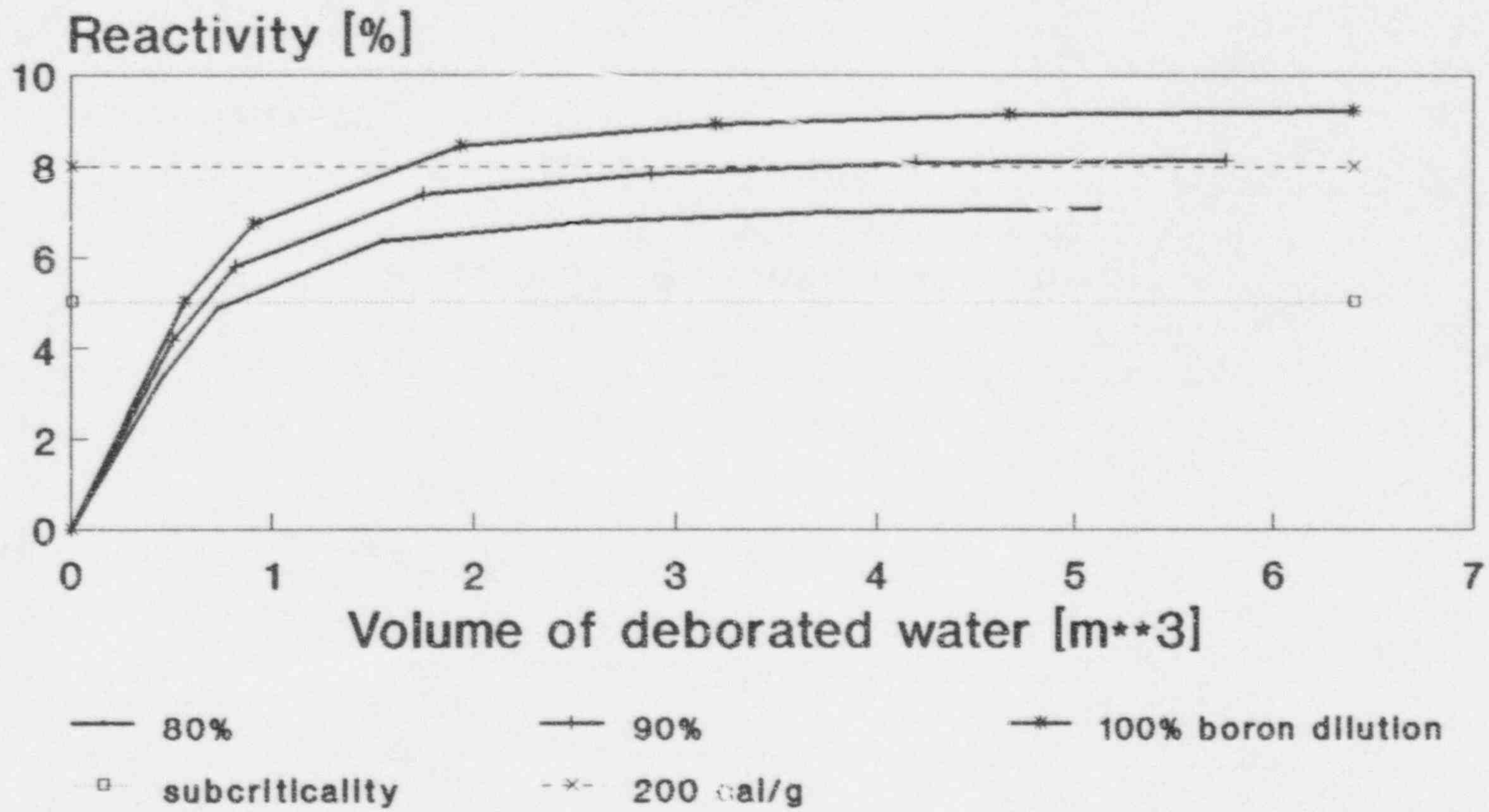


Fig. 3 Asymmetrical boron dilution
hot zero power
critical boron concentration: 1082ppm

STATIONARY 2D CORE CALCULATION WITH QUABOX/CUBBOX

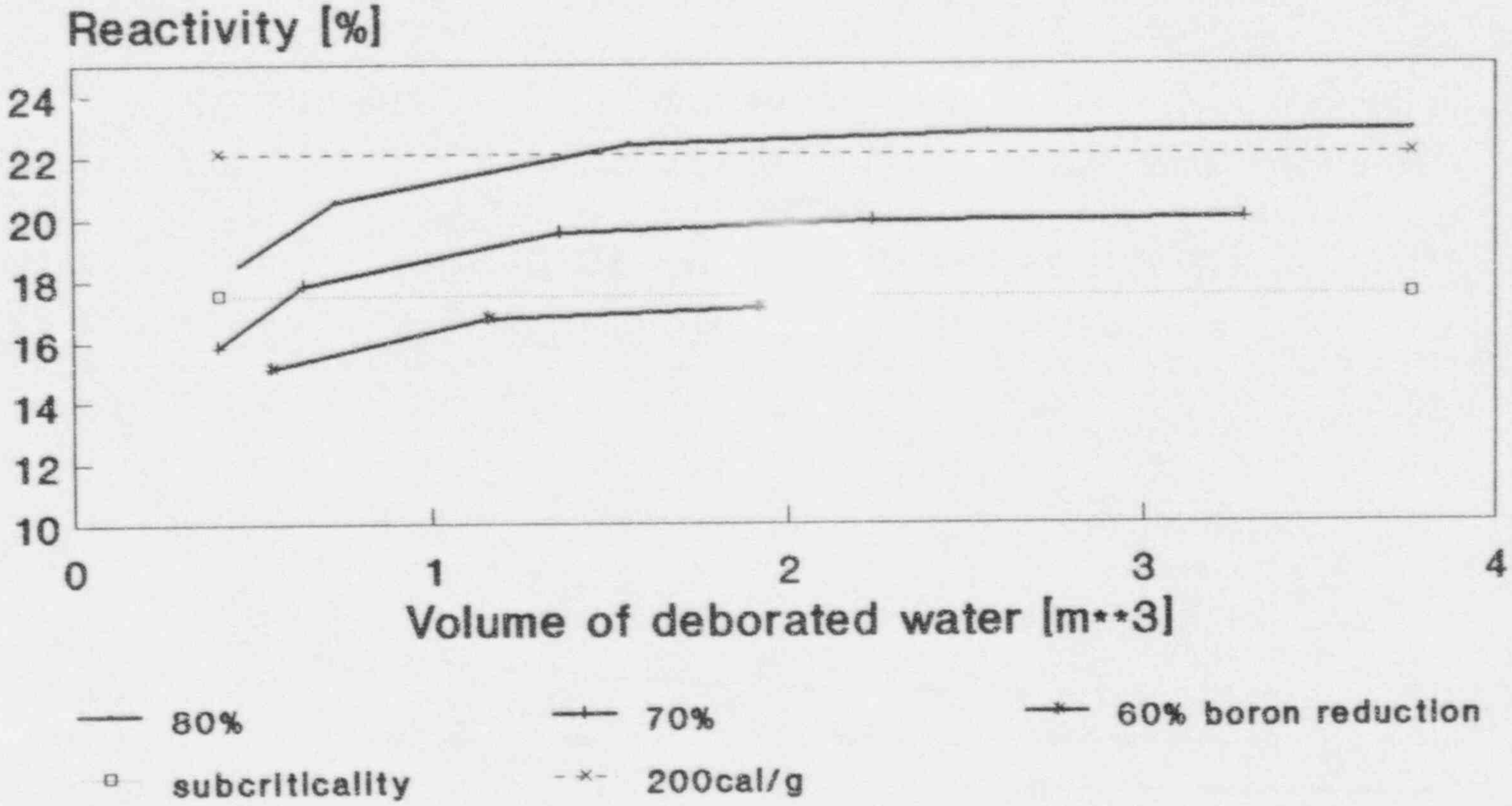


Fig. 4 Asymmetrical boron dilution
cold reactor state
initial boron concentration: 2200ppm

OECD / NEA / CSNI Specialist Meeting on
Boron Dilution Reactivity Transients
State College, Pennsylvania, USA, 18-20 October 1995

SCOPING STUDY OF INHERENT BORON DILUTION FOR LOVIISA NUCLEAR POWER PLANT

Heikki Kantee
IVO International Ltd, Rajatorpantie 8 Vantaa, 01019 IVO, Finland
Samuli Savolainen
Loviisa Nuclear Power Plant, P.O.Box 23, 07901 Loviisa, Finland

INTRODUCTION

Boron dilution analyses have recently become one of the most interesting topics concerning PWR safety in Finland. Traditionally the dilution analyses are mainly dealing with external dilution mechanisms but in 1993 the Finnish Centre for Radiation and Nuclear Safety required Imatran Voima Oy to carry out a wide study concerning all possible nonhomogeneous boron dilution mechanisms, consequences and their mitigation.

In dilution analysis it has traditionally been assumed that diluted coolant is injected into the primary circuit from an external source. Mitigation against these events is normally straight forward. The probability of inadvertent external dilution can be reduced by preventive automation. For inherent nonhomogeneous boron dilution events the problem is more difficult. The dilution processes are strongly dependent on initial events and system behavior. Additionally the slug formation and its interaction is complicated to predict accurately.

The study was initiated by performing deterministic analyses of physical phenomena leading to inherent boron dilution. These analyses were then used as a boundary conditions in final reactor-dynamic analyses. Moreover, mixing analyses were needed to get a realistic boron concentration at the core inlet.

The object of this paper is to discuss the thermal-hydraulic scoping analyses which were done to study the inherent boron dilution mechanisms. Mixing and reactor-dynamic analyses, which are also needed to determine the core response, are not discussed in this context.

The boiling-condensing heat transfer mode is the most significant precondition for inherent boron dilution [1, 2]. E.g. during a small break loss of coolant accident (SBLOCA) secondary side cooling is needed to get the reactor to the cold shut-down stage. This emergency operation may lead to boiling condensing heat transfer mode. In primary-to-secondary (PRISE) leakage accident there is a possibility that leak flow reverses as a consequence of operator action and may cause a boron dilution problem. The whole study was separated into three categories:

-Scoping study of possible accidents and transients leading to boiling-condensing mode

-Mixing processes

-Reactor-dynamic consequences

The major attention of the scoping studies was paid to the following initial events:

-SBLOCA

-PRISE

-ATWS (Anticipated Transient Without Scram)

In addition the study covers also other conditions like:

-Accidents during outage

-Initial events leading to severe accidents (e.g. total loss-of-feedwater)

The main tool in thermal-hydraulic analyses was RELAP5/MOD3.1 [3] computer code. Because the capability of the code is limited to trace boron concentration in different parts of the primary circuit, main attention was paid on steam condensation in steam generators or reverse flow in PRISE accidents. Mixing of low boron concentration water slug with high boron concentration coolant on the way to the core was left to be calculated with codes having this capability.

BORON DILUTION PHENOMENON IN PRIMARY CIRCUIT

Inherent boron dilution phenomenon can be expected when during accident or transient primary coolant level in the reactor pressure vessel (RPV) decreases below hot leg elevation and as a result steam can flow into the steam generators. Because boron solute is in practice transported only in the liquid phase steam can be assumed to have zero boron concentration. In Loviisa, like in all PWRs, the cooldown of primary circuit during accident is first carried out by removing residual heat via steam generators by cooling down secondary side. During this phase steam condenses in the steam generator primary side and as a consequence a water slug having a low or in the worst case zero boron concentration may be formed in the cold leg loop seal. The longer this boiling-condensing mode continues, the lower boron concentration in the loop seal becomes. Then the low boron water slug may for some reason begin to flow towards the pressure vessel. Depending on into which loops the high pressure injection (HPI) takes place boron concentration in the cold leg may increase due to mixing with HPI water before entering RPV. If there is no injection into the loop concerned, boron concentration of the water slug may increase only in downcomer and in lower plenum before entering the core. Some part of the water slug may also flow to the break depending on the break location. When the water slug finally enters the core power behavior depends very much on boron concentration, temperature and velocity of the water slug. The development on the inherent boron dilution phenomenon e.g. during LOCA can be divided roughly into six phases:

1) Initiating event

- Subcooling margin is lost
- Residual heat removal is disturbed

2) RPV liquid level decreases below hot leg elevation

- Steam flows into hot legs and further into steam generators

3) Primary circuit cooling via steam generators

- Steam condenses in steam generator and forms a low boron concentration water slug first in the cold leg loop seal

4) Water slug flows to the RPV when natural circulation restarts

- Mixing with the ECC water on the way to the RPV
- Depending on break location some portion of the slug out of the break

5) Mixing in the RPV

- Temperature difference between slug and water in the RPV
- Pressure below low pressure injection pumps shut-off head

6) Water slug enters the core

- Boron concentration
- Temperature
- Velocity

The initiator to push the low boron concentration water slug to the RPV can e.g. be natural circulation restart as a result of liquid level rise above hot leg elevation. In case of Loviisa this usually means that primary coolant inventory has increased enough as a result of increased ECC injection. Another way to start slug movement is perhaps caused by the operator when opening the vent valves which are used to remove noncondensable gases from the primary circuit. These gases are formed during accident, especially as a result of accumulator injection.

The final core response is not discussed in this paper since RELAP5/MOD3.1 is not capable of calculating in detail enough the behavior of those parameters which were presented above in item 6). Results from RELAP5/MOD3.1 analyses are to be used as boundary conditions in the analyses which are carried out using the code like PHOENICS [4], HEXTRAN [5] and SMATRA [6].

SMALL BREAK LOCA ANALYSIS

Since no earlier data was available concerning boron dilution in connection with SBLOCA (100 % power level), a large number of accidents with different break sizes were analyzed. Both hot and cold leg breaks were considered including also stuck open pressurizer safety valve case as a special hot leg break. Break size was varied from 5 cm^2 to 30 cm^2 with 5 cm^2 increment in cold leg breaks and from 20 cm^2 to 40 cm^2 with the same increment in hot leg breaks. Also starting time of operator action was varied. The fastest action time was 5 min after the initiation of the accident and the slowest 30 min. Third varied parameter was the emergency operating procedure (EOP) the operators will follow, i.e. are they going to follow EOP for large (LBLOCA) or SBLOCA. SBLOCA EOP is meant for leakages so small that make-up pumps or at least high pressure injection (HPI) can maintain primary coolant subcooling margin. The break sizes considered here are, however, so "large" that operators should follow LBLOCA EOP.

Based on Loviisa configuration a situation where only one HPI pump is available can be postulated during a cold leg break if a single failure assumption is considered. This is possible if break is assumed to locate in such a way that the injection of both HPI pumps of one redundancy is lost to the break and as a result of a diesel generator starting failure one HPI pump in another redundancy is not available. Boron and make-up injection, which in Loviisa takes place into hot leg, is assumed to be lost via break in the hot leg break cases but not in the cold leg break cases. If single failure assumption is followed in the hot leg break accidents only one HPI pump is lost. However, hot leg break analyses were carried out with assumptions of both three and only one HPI pump available. This was due to the fact that with three HPI pumps available no boron dilution problem during hot leg break accidents was discovered.

One way to find out which cases may cause boron dilution problems is to calculate total amount of steam which condenses in the steam generators. The amount shows how much non-boric liquid is formed during an accident since there is not any other place in the primary circuit where major condensation can take place. Fig. 1 shows the total condensation in each six steam generator during the case in which largest amount of steam was condensed. It can be seen that condensed amount differs between each steam generator. This means that in the nodalization model it is very important to model each loop separately since loops are in different positions from the boron dilution point of view. A considerable amount of steam was condensed between 3000 s and 4600 s. Reason for this is that during this period liquid level in pressure vessel remains below hot leg elevation which is in turn due to the fact that accumulators are empty at 2800 s and primary pressure decreases below shut-off head of low pressure injection pumps at 4000 s. The length of this period was found to be an essential parameter in steam condensing phenomenon. Although the total condensed amount in this case is considerable, one should remember that part of the condensation takes place under two-phase flow conditions, mixing takes place with HPI water in the cold leg and with coolant having high boron concentration in pressure vessel. In cold leg break cases part of the diluted water may also flow from the downcomer to the break.

In Fig. 2 boron concentration behavior in each cold leg loop seal is presented. As one would expect the boron concentration behavior is very closely related to the condensed amount of steam in steam generators.

21 different SBLOCA parameter variations were calculated altogether and following conclusions were drawn:

1) Cold leg breaks cause clearly bigger boron dilution potential than hot leg breaks even though only one HPI pump and neither make-up nor boron injection is assumed in hot leg breaks. This is mainly due to the fact that break "steals" a considerable amount of steam which in cold leg case flows to the steam generators.

2) Fast operator action reduces the dilution potential.

3) Presently followed EOP (LBLOCA) mitigates boron dilution problem. If the operators follow SBLOCA EOP (slow secondary side cooldown), accident proceeds slower and there is more potential for boron dilution.

PRIMARY-TO-SECONDARY SIDE LEAKAGES

During PRISE leakages one of the operator's main task is to decrease primary pressure rapidly below the opening set point of the steam generator safety valve. In this way the possibility to have a containment bypass leakage is reduced. On the other hand this kind of action may result in the situation where break flow reverses and secondary side water flows to the primary circuit. The situation becomes even worse if feedwater isolation fails. In Loviisa primary pressure degradation is realized by spraying the pressurizer and cooling down primary coolant via intact steam generators. The defect steam generator is isolated in the early phase of the PRISE accident.

In traditional PRISE analyses the emergency core cooling (ECC) and make-up systems are assumed to operate at their maximum capacity. When boron dilution problem is analyzed the situation is vice versa since ECC and make-up injections tend to resist primary pressure reduction in the early phase of the accident and thus with maximum capacity it takes longer time until the flow can reverse. This increases boron concentration in the defect steam generator secondary side and when or if flow finally reverses the influence on dilution is smaller. Another feature in traditional analyses was that defect steam generator safety valve was assumed to stick to open position, e.g. when water begins to flow through it [7]. With this kind of assumption flow reversal in the early phase would be very unlike and that is why safety valve was assumed to operate as designed. The starting time of operator action plays also important role since flow reversal without operator action is almost impossible. Only defect steam generator isolation is automatic; pressurizer spraying and cooling via intact steam generators are done by the operators.

As a general observation it can be mentioned that in case of primary collector cover break, which is considered as a maximum break size in Loviisa, the flow reversal does not take place until defect steam generator has become water solid (filled up with primary coolant). This means that large amount of warmer primary coolant has flown into the steam generator and boron concentration there is considerable even if ideal mixing is assumed. If the location of break is on the tube area the secondary side boron concentration may in this case be lower at the flow reversal moment but because of the break area is smaller the reverse mass flow rate is also smaller. A single tube rupture was considered too small leakage and it was not analyzed.

During the case where the largest backflow was predicted about 2.7 tons flew from defect steam generator to the primary circuit between 200 s and 500 s (Fig. 3). However, before flow reversed a considerable amount of warmer primary coolant had flown to the defect steam generator (approx. 29 tons). Thus the reverse flow was mixture of primary and secondary water and did not have any drastic effect on boron concentration in the defect loop cold leg (Fig. 4, Note: Lops YA12, YA15 and YA16 behave very much in the same way). The calculation was stopped at 2000 s though some reversed flow could be still seen. However, nothing dramatic can be expected during the rest of the time when reactor would be slowly taken down to cold shut down.

Eight different cases were analyzed dealing with large and medium size PRISE leakages (100 % power level). HPI pumps from one redundancy were assumed not to be available. The parameters varied were initiation of operator action, pressurizer spraying capacity and intact steam generators cooling gradient. The following conclusions were drawn:

1) Large PRISE leakage results in more unfavorable situation than the smaller ones

2) Rapid operator action (at about 2.5 min in large PRISE) leads to maximum flow reversal

3) PRISE accident does not lead to any major safety problem **if feedwater isolation from defect steam generator does not fail**

4) PRISE accident is a special case for SBLOCA. However, boiling-condensing phenomenon could not be seen in the intact steam generators during secondary side cooldown.

It should be emphasized that the isolation of all feed water is very important since otherwise there will an external source of non-boric water to decrease boron concentration in the defect loop. In Loviisa this is implemented by isolating steam generator of main feed water from high steam generator level signal. There are two isolation valves in each steam generator and in addition to this, of course, a control valve which also will close from high level.

ANTICIPATED TRANSIENTS WITHOUT SCRAM

It is good to remember that ATWS cases form a group of accidents which have a special feature in boron dilution problem, i.e. the core can tolerate very limited amount of diluted water since the control rods are not in the core. This means that boiling-condensing mode must be avoided by trying to inject boron into the primary circuit using the pumps with high shut-off head. Pressure reduction via secondary side cooling may cause a considerable power excursion. This together with a possible diluted slug entering the core may lead even to significant core damages or primary pressure may exceed the acceptance criterion.

The sticking mechanism of the control rods during ATWS events can be roughly divided into two categories. First, the rods may stick as a consequence of signal or electrical failure. In these cases it could be assumed that the operator is able to cut the electricity from the magnets holding the control rods above the core before the slug is formed and especially before it enters the core. After the rods have fallen into the core, the core can tolerate more dilution. The second mechanism is mechanical in which the rods movement into the core is prevented by some mechanical failure. In this kind of situation it could be assumed that only a limited number of adjacent rods will stick and remain above the core for the reason of some common cause failure. The rest of the rods will drop into the core after the scram signal. These assumptions were considered also in this study as parameter variations and they really had a positive influence on the results. All the analyses expect those parameter studies were, however, done assuming that all the control rods will remain at the position they were on the moment the calculation was started.

Two different initiating events has been analyzed so far:

1) Loss-of-offsite power (LOOP) from 100 % power level

2) Control rod group withdrawal from 1 % power level (CRWD)

More 40 parameter variations concerning LOOP were calculated using codes like HEXTRAN, SMATRA and even RELAP5/MOD3.1. HEXTRAN code includes 3-dimensional reactor kinetics model, SMATRA has 1-dimensional model and RELAP5 point model. Coolant density reactivity feedback was found to play important role in LOOP events and the beginning of cycle was found to give worse results than the same event later during cycle. The analyses also clearly showed that even smaller changes in boron concentration could have significant effect on core power behavior.

Results from one LOOP case are presented in Fig. 5 and 6. This event was calculated with HEXTRAN code. As we can see from Fig. 5 at about 900 s the fission power decreases down to practically zero and one might think that the situation is now over since everything seems to be "nice and steady". However, the truth is that the situation after 900 s is very sensitive since natural circulation has stopped (Fig. 6) because the level in RPV has decreased below hot leg elevation and fission power has "died" because of boiling in the core. The behavior of fission power and primary circuit flow parameters indicate that the boiling-condensing mode is on and the operators should be very careful when deciding the future measures. Cooling down primary circuit via secondary side should be avoided as much as possible to prevent natural circulation from restarting before the control rods have been inserted.

Only preliminary results are at the moment available concerning CRWD events and based on those it can be concluded that one of the most important parameter seems to be the initial location of the control rod group which is withdrawn. Concerning the stagnation of natural circulation the worst case is not the one when the control rod group is initially deepest in the core but somewhere on the middle elevation. This information is based RELAP5/MOD3.1 analyses and it will be checked using HEXTRAN code.

The results from both initiating events show that formation of boron diluted water slugs cannot be ignored. One of the most important parameter is the amount of primary coolant lost through the pressurizer safety valve. If the level in the pressure vessel decreases down to the hot leg elevation, single phase natural circulation is interrupted. Thereafter steam may flow into the steam generators and condense there forming the water slug with low boron concentration. Later on single phase natural circulation may restart and push the water slug into the core. The results also show that proceeding of ATWS accident is sensitive to boundary conditions, system behavior and reactivity coefficients. Also steam generator nodalization model plays some role at least in Loviisa type horizontal steam generator and too rough model should be avoided.

So far the final analyses concerning the power response when the slug enters the core have not been calculated. These analyses will be, however, carried out in the near future using the HEXTRAN code.

TOTAL LOSS-OF-FEEDWATER ACCIDENTS

Accidents discussed here are beyond design basis accidents. No operator action was assumed until the cooling via secondary side was initiated by starting the emergency or auxiliary emergency feedwater injection. The main objective of these analyses was to find out how long time it takes until:

1) Steam generators dry out

2) Liquid level in the pressure vessel decreases below the hot leg elevation

3) Core heat-up begins

From the inherent boron dilution point of view the most interesting item in the above list is 2), because after that point of time boiling-condensing becomes effective. This means that the operators should pay attention on the measures they are choosing if they are able to restart feedwater injection after that point of time. Reducing the primary pressure by cooling down steam generators should be seriously considered and e.g. primary "bleed & feed" may be safer way to reduce primary pressure below HPI pumps shut-off head.

Analyses dealt with two different initial events. The first one was total loss-of-feedwater (LOFW) in such a way that all main, emergency and auxiliary emergency feedwater pumps were lost. Everything else operated as designed. The second case was total loss of off-site AC power including emergency diesel generators (BLACKOUT). The main difference between these two cases was the point of time at which the three above mentioned phenomena were predicted. In the first case the accident proceeded faster since steam generators dry-out happened earlier. This was due to the fact that reactor coolant pumps were running until level in steam generators decreased below pumps stop set point and during this period a lot secondary side water inventory was lost due to strong boiling. In BLACKOUT reactor coolant pumps stopped quickly since the electric power was lost. The difference between average dry-out (did not take place at the same time in all steam generators) time was approximately 2 h. However, item 2) was predicted to take place in BLACKOUT case somewhat earlier (at 5.8 h) because in LOFW case there was a period during which primary pressure decreased below HPI shut-off head. As a consequence primary coolant inventory increased and item 2) was postponed to 6.5 h. Item 3) happens shortly after the liquid level reaches the top of the core. In BLACKOUT case it was assumed that operators manage to start diesel generators just at the last minute before core heat-up would begin (9.5 h). In LOFW case one auxiliary emergency feedwater pump was assumed to start also at the last moment (8 h). These assumptions result in situations where injection takes place either to all six steam generators using two emergency feedwater pumps or only to four steam generators with one auxiliary emergency feedwater pump.

Based on more effective cooling via secondary side much more steam was condensed in steam generators during the BLACKOUT case (approx. 44 tons) than in the LOFW case (approx. 17 tons). This naturally resulted in considerable lower cold leg boron concentration in BLACKOUT case.

CONCLUSIONS

The results of an extensive scoping study of inherent boron dilution analyses for Loviisa NPP concerning initiating events of SBLOCA, PRISE, ATWS and total loss-of-feedwater cases has been presented. These events were considered to be the major sources for inherent formation of pure or low boron concentration water slugs.

The results show, excluding PRISE if feedwater isolation is not failed, that there is a chance during these accidents for low boron water slug formation in the cold leg loop seal. How the situation develops after the re-establishment of natural circulation cannot generally be predicted with thermal-hydraulic system analysis codes. The boron concentration of the water slug will most probably increase on the way to the core due to e.g. high pressure injection in the cold leg and mixing with water having high boron concentration and different temperature in the pressure vessel downcomer and lower plenum. System analysis codes are not capable of calculating mixing, since this is a 3D process including turbulent mixing as a main diffusion phenomenon. One way to avoid this problem is to conservatively assume that the water slug enters the core without any mixing on the way. This can be applied as a limiting case as long as the water slug does have high enough boron concentration not to cause even recriticality in the core. If this is not the case then one has to evaluate the degree of mixing and finally show that the reactor dynamical consequences are acceptable. That is why mixing analyses using PHOENICS code has also been carried out during this study. The results are not discussed in this paper since there will be a separate paper [8] in this meeting in which those analyses are discussed in more detail. The reactor dynamical consequences are calculated with code called HEXTRAN including the three dimensional kinetics calculation.

REFERENCES

[1] J. Hyvärinen, An Inherent Boron Dilution Mechanism in Pressurized Water Reactors, The Fifth International Topical Meeting on Reactor Thermal Hydraulics, NURETH-5, Salt Lake City, Utah, USA, September 21-24, 1992

[2] J. Hyvärinen, An Inherent Boron Dilution Mechanism in Pressurized Water Reactors, Nuclear Engineering and Design 145 (1993), pp. 227-240.

[3] K.E.Carlson et al., RELAP5/MOD3 Code Manual, Volume 1: Code Structure, System Models and Solution Methods (Draft), NUREG/CR-5535, EGG-2596, June 1990

[4] D.B.Spalding, A Guide to PHOENICS Input Language, CHAM/TR100, 16/07/92

[5] R. Kyrki-Rajamäki, HEXTRAN: Three Dimensional Reactor Dynamics Code for VVER Reactor Cores, Proceedings of the International Topical Meeting on Advances in Mathematics, Computations and Reactor Physics, Pittsburgh, VA, Apr. 4 - May 5, 1991, American Nuclear Society, LaGrange Park, Ill, 30.2 4-1 - 30.2 4-5.

[6] R. Kyrki-Rajamäki et al., Validation of SMATRA Accident Analysis Code against Loviisa Plant Data, Proceedings of the 3rd International Symposium on Power Plant Transients - 1992 ASME Winter Annual Meeting, Anaheim, CA, Nov. 8 - 13, American Society of Mechanical Engineers, New York, NY, FED-Vol. 140, 153 - 162

[7] H. Kantee and H. Tuomisto, Large Primary-to-Secondary Leakage Accidents in Loviisa VVER-440 Reactor, RELAP5 International Users Seminar, Boston, Massachusetts, USA, July 6-9, 1993

[8] P. Gango, Application of Numerical Modelling for Studying Boron Mixing in Loviisa NPP, CSNI Specialist Meeting on Boron Dilution Reactivity Transients, State College, PA, October 18 - 20, 1995

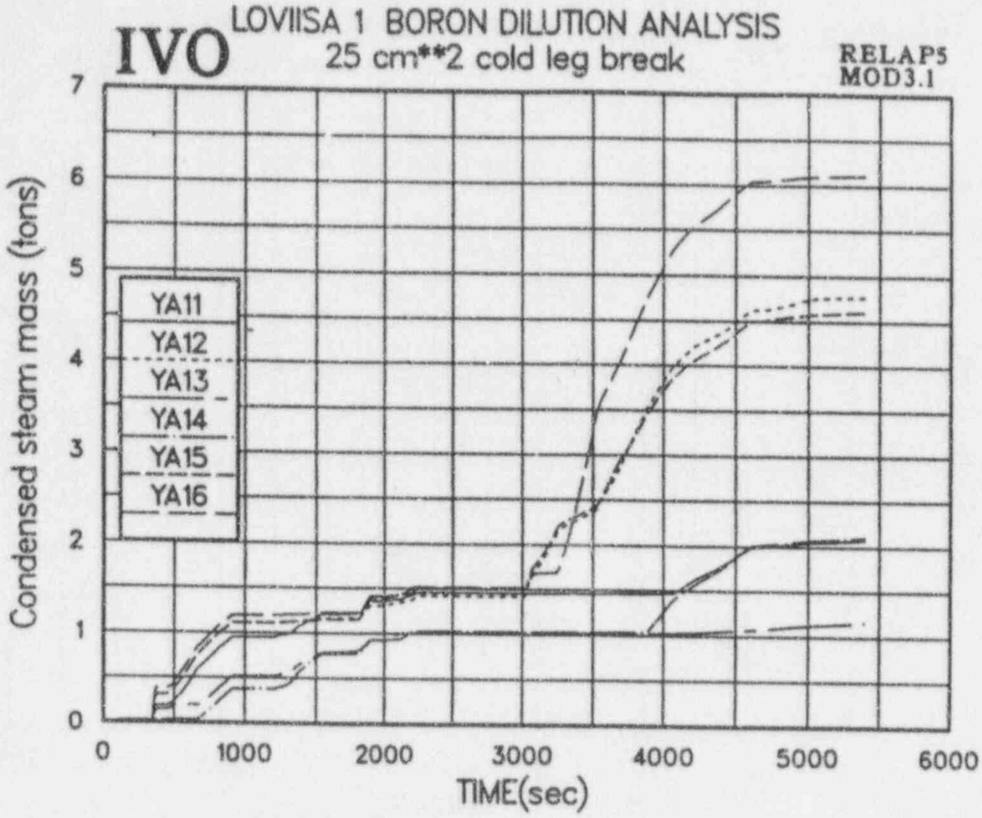


Fig. 1 Total condensation in each six steam generator

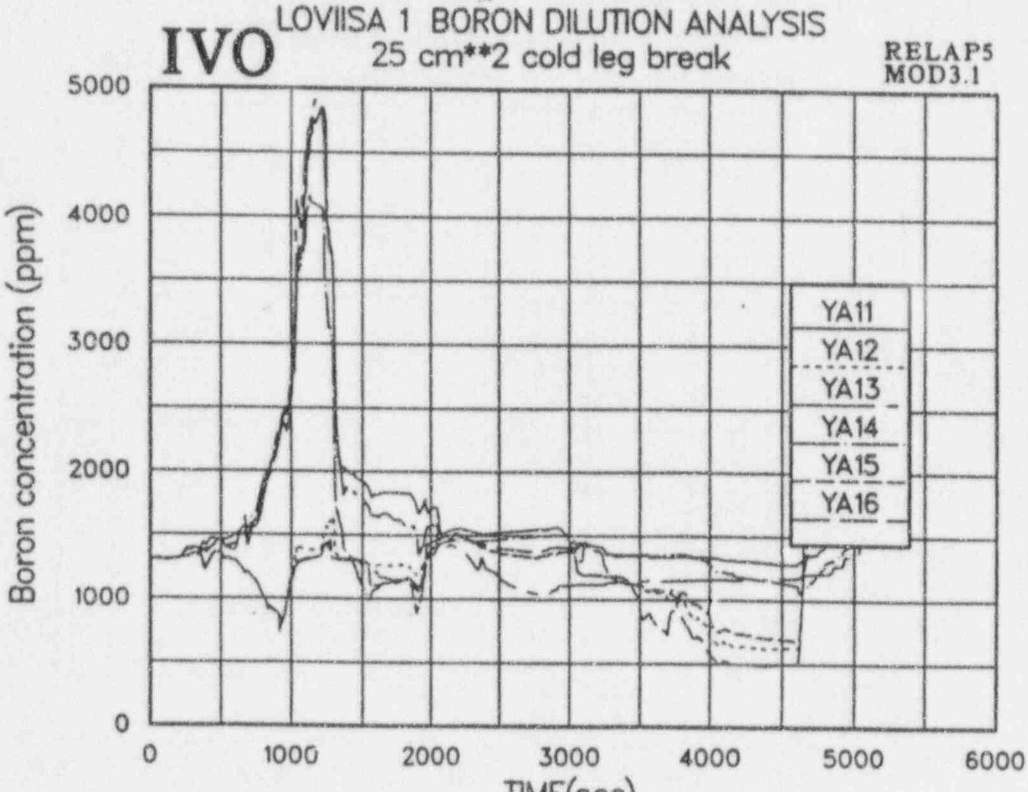


Fig. 2 Boron concentration behavior in each six cold leg loop seal

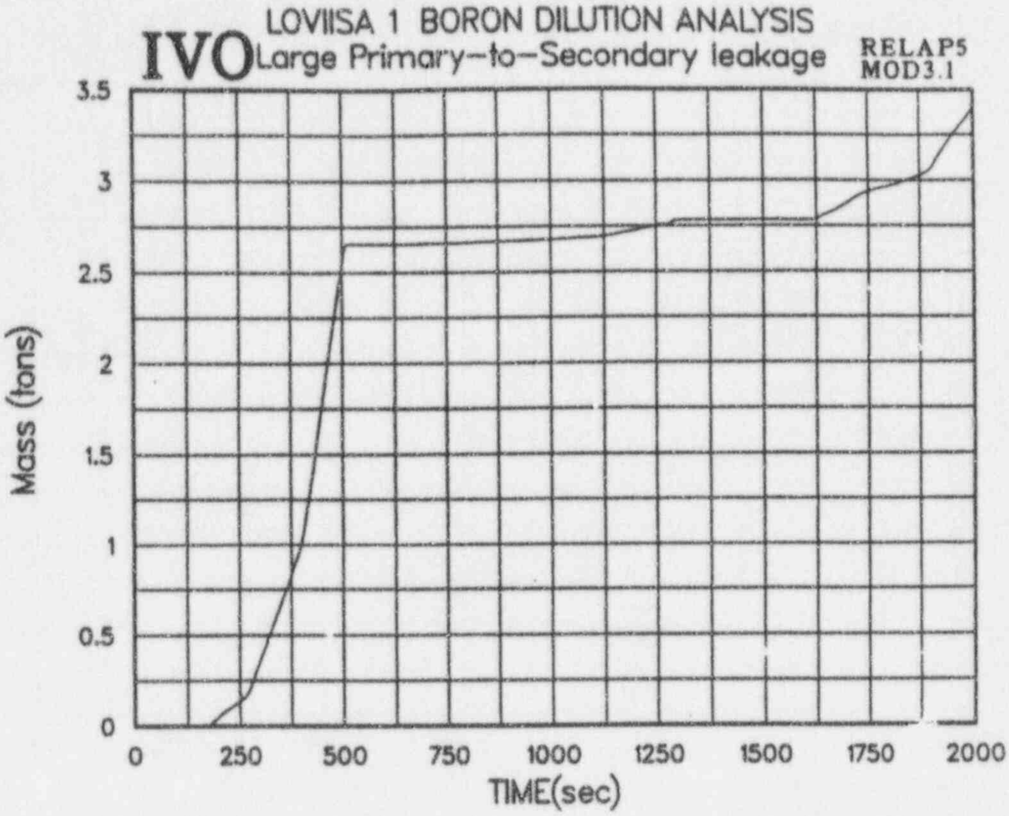


Fig. 3 Integrated flow from defect steam generator to the primary circuit

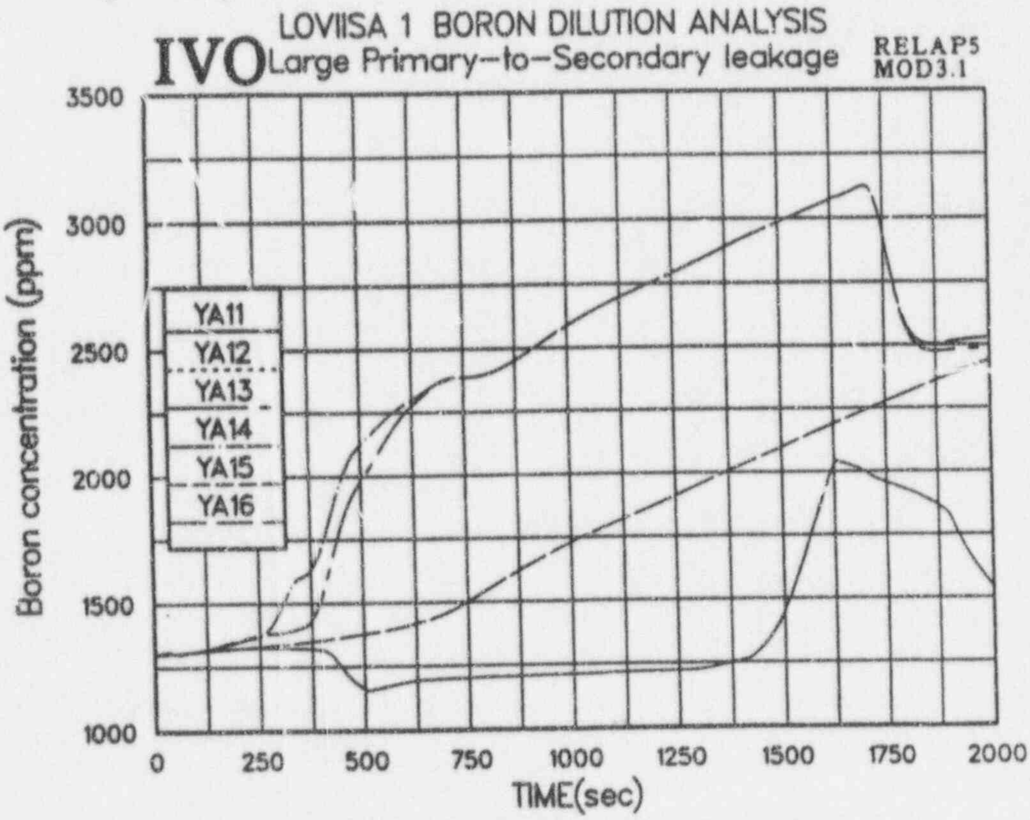


Fig. 4 Boron concentration behavior in the cold leg loop seal

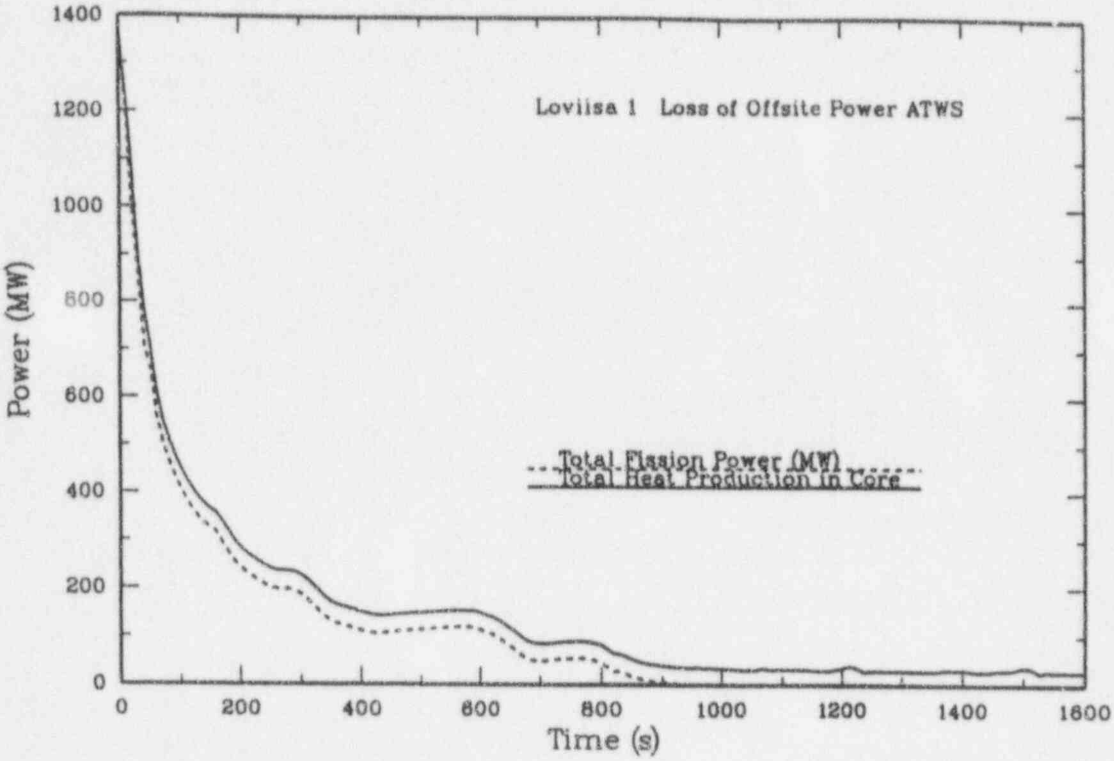


Fig. 5 Core power behavior

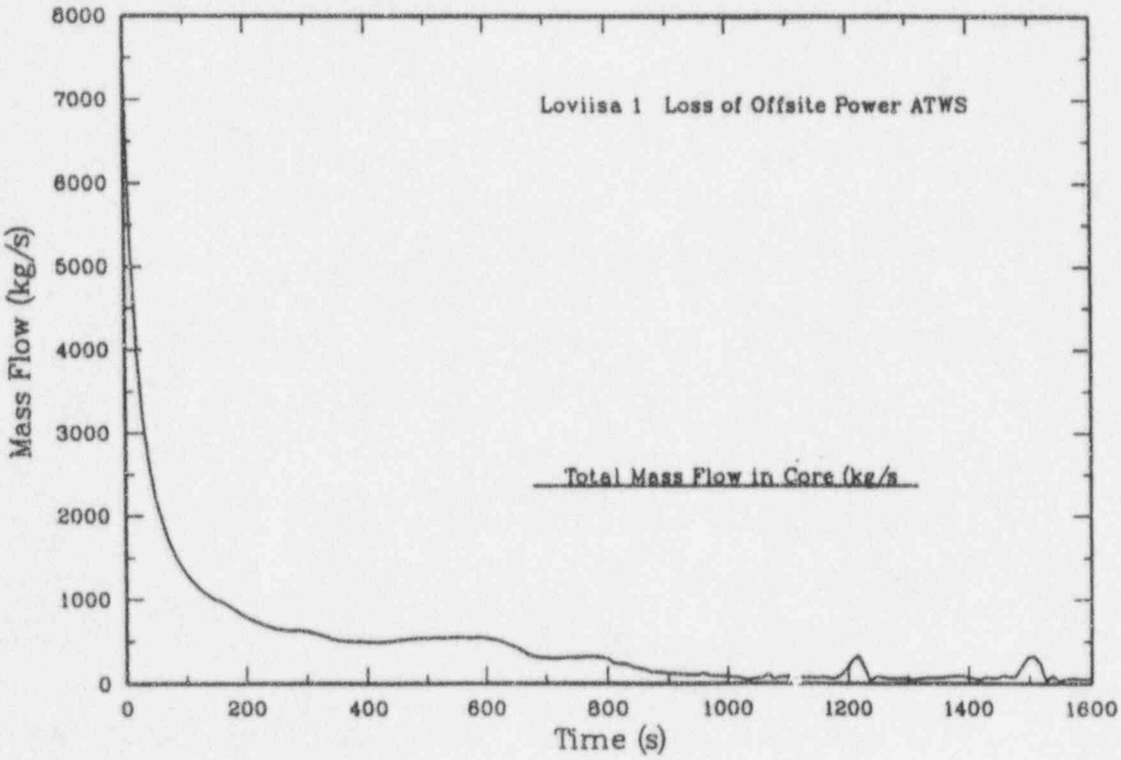


Fig. 6 Core entrance mass flow rate behavior

OECD / NEA / CSNI Specialist Meeting on
Boron Dilution Reactivity Transients
State College, Pennsylvania, U.S.A., 18-20 October 1995

ANALYSIS OF CORE RESPONSE TO THE INJECTION OF DILUTED SLUGS FOR THE LOVIISA VVER-440 REACTORS

P.Siltanen, M.Antila
IVO International Ltd, Rajatorpantie 8 Vantaa, 01019 IVO, Finland
R.Kyrki-Rajamäki, T.Vanttola
VTT Energy, Nuclear Energy, P.O.Box 1604, 02044 VTT, Finland

Abstract

During the past several years major attention has been given to analyzing the possibilities and potential consequences of a slug of diluted water entering the core of the VVER-440 reactors in Loviisa. Suitable countermeasures have already been implemented to reduce the probability of external inhomogeneous dilution. Inhomogeneous dilution scenarios with their high reactivity disturbance potential were not included in the original safety analyses. Typical dilution scenarios are introduced.. The paper discusses further both the static reactivity potential of diluted slugs and the results of dynamic analysis of core response under various conditions. The main purpose of these analyses has been to map the limiting conditions for the onset of severe core damage in these reactivity accidents. The principles of the new preventive automation recently implemented at Loviisa NPS are also described.

1. INTRODUCTION

During recent years major attention has been given to analyzing the possibilities and potential consequences of a slug of diluted water entering the reactor core [1,2]. Similar analyses have been performed for the VVER-440 reactors in Loviisa [3,4]. Suitable countermeasures have already been implemented to reduce the probability of external inhomogeneous dilution. The possibility and consequences of inherent boron dilution mechanisms [5] during Small Break Loss of Coolant Accidents (SBLOCA), Steam Generator Tube Ruptures (SGTR) and Anticipated Transients Without Scram (ATWS) are also being analyzed. Inhomogeneous dilution scenarios with their high reactivity disturbance potential were not included in the original safety analyses, which are now being updated.

The cases analyzed in this paper pertain to external dilution scenarios where a slug of diluted water is assumed to be formed in a stagnant reactor coolant loop. Stagnant flow conditions can be created in a VVER-440 reactor under three somewhat different circumstances. First, a main gate valve can be closed while the reactor coolant pumps in other loops are operating. Erroneous startup of the loop could drive a slug of diluted water into a sector of the core. Secondly, the reactor can enter a condition of weak and unstable natural circulation when all reactor coolant pumps stop simultaneously, e.g. during startup dilution. A diluted slug could then be driven into the core by natural circulation or by the restart of the first reactor coolant pump. Thirdly, the reactor can be in a shutdown condition with natural circulation cooling via one or two loops and with the rest of the loops isolated. A diluted slug resulting e.g. from maintenance work could again be driven into the core by natural circulation or by the restart of the first reactor coolant pump.

The purpose of analyzing the core response to diluted slugs is to obtain information on the potential for severe fuel damage associated with diluted slug volume and geometry, its boron disturbance and the speed of injection into the core. In this respect a useful measure of the diluted slug is the static reactivity potential of the slug in the core, disregarding any nuclear feedback effects. The calculations reported in this paper provide information on the relationship between the static reactivity worth and the actual consequences.

Results of potential consequences in the core for three typical external dilution scenarios in the hot reactor condition are presented. The calculations have been performed by our detailed three-dimensional core dynamics model HEXTRAN coupled with an extensive thermal-hydraulic circuit model SMABRE.

The principles of the new preventive automation recently implemented at Loviisa NPS are also described.

2. DILUTION SCENARIOS IN HOT CONDITIONS

Figure 1 provides information on the layout of the primary circuit and on the injection points of the make-up water system (TC) for the Loviisa reactors. A special feature of Loviisa is that the injection lines for make-up water are open only into three loops out of six, both into the hot (TC10) and cold legs (TC50). Similarly, let-down lines are open only from the three remaining loops. The purpose of this arrangement is to enhance the separation of the high pressure safety injection (TJ) paths. The arrangement influences how the injection flows are distributed into different loops.

The following are the most important dilution scenarios for hot reactor conditions.

2.1 Closing of a main gate valve

The reactor is assumed to be close to criticality during start-up dilution or to be operating at partial power. The control rods are withdrawn from the core. Reactor trip does not occur when at least three reactor coolant pumps are in operation. The main gate valve in the cold leg of a loop may be temporarily closed due to failure of an anti-rotation device or maintenance work on the pump motor. The coolant in the loop can become stagnant, particularly in three of the loops without reverse flow through the TC system pipelines. Simultaneously, the make-up water lines into these loops are open. The full injection capacity has a potential for some of the dilution water to enter the hot leg of the closed loop. This point is illustrated in Figure 2 on a pressure level scheme for the different injection points and for the make-up system common collector pipelines. It is also conceivable that pure condensate left in the injection pipelines during preceding dilution could enter the loop.

The slug of diluted water could be transported to the core if the reactor coolant pump is started before opening the main gate valve. The probability of this transient scenario is estimated to be < $3*10^{-7}$ per year taking into account the new preventive automation.

2.2 Stopping of all reactor coolant pumps during start-up dilution

The reactor is assumed to be close to criticality during start-up dilution or to be operating at power when all the reactor coolant pumps stop. Reactor trip is activated and the reactor becomes subcritical with inserted control rods. Natural circulation with all main gate valves open takes over. Particularly during start-up after refuelling or after a longer outage the decay heat power of the core is small. Flow stagnation may occur in some loops. Dilution of the primary circuit after the pumps have come to a stand-still may create a slug of diluted water in one or more of the loops with open injection lines. Pure condensate or diluted water left in the injection pipelines during preceding dilution may also be pushed into the loops by continued make-up water supply.

The make-up system pumps are considered to operate rather independently of the reactor coolant pumps. There are two main reasons for this. The reactor coolant pumps can stop for reasons other than just loss of external power and the make-up system pumps are secured with emergency power.

A slug of diluted water in a loop can later be transported to the core if the pump in this loop is the first one to be started. It is conceivable that restart of natural circulation could also transport the slug into the reactor. This can occur without considerable mixing only if the slug is relatively isothermal with the rest of the water in the loop. In this case the slug might reach the reactor vessel before the restart of any coolant pumps. The probability of this scenario was estimated to be < $5*10^{-7}$ per year taking into account the new preventive automation.

2.3 Accident conditions

In SBLOCA and ATWS events the inherent formation of a diluted slug in the loop seals is possible by a boiler/condenser cooling mode if the primary circuit coolant inventory is reduced sufficiently to stop the two phase natural circulation. During SGTR accident diluted water might flow to the primary circuit from the secondary side of the damaged steam generator. The slug may later enter the core as a result of restart of natural circulation or of a reactor coolant pump [5]. These scenarios are not discussed further in this paper, but experience in analyzing these events indicates that ATWS events at beginning of cycle are of some concern. The most adverse event is the inadvertent withdrawal of control rods from low power.

3. REACTIVITY EFFECTS OF DILUTED SLUGS

The geometry and the degree of mixing with which a diluted slug finally enters the core has a significant influence on the amount of positive reactivity inserted into the core. For the same integrated boron disturbance in the core, the most reactive configuration is a compact unmixed slug, which produces a strong local peaking of the neutron flux.

Mixing of the diluted slug with non-diluted coolant along the path from the origin of the slug up to the reactor core provides a mechanism for effective inherent protection against small local diluted slugs particularly in the case of natural circulation, if density differences exist. On the other hand, mixing is known to be rather weak in the case of forced steady state flow when the flow from different loops is channeled into sectors in the downcomer and reactor lower plenum. More mixing can be expected during the restart of the first coolant pump.

Some numerical studies of mixing phenomena have been carried out for the Loviisa reactors using the PHOENICS flow calculation package [6]. Numerical diffusion, large problem size and lack of experimental validation data are the main concerns. The lack of reliable information on coolant mixing can be compensated by using conservative approximations in the first place when analyzing core responses, as is the case in this paper.

A useful measure of the size, shape and boron disturbance of a diluted slug is provided by its maximum static reactivity potential in the core, disregarding any nuclear feedback effects. This reactivity insertion into the core would be obtained if the slug were to enter the core at high velocity in less than one second. This is the case at full flow rate with six reactor coolant pumps in operation. In a rapid reactivity insertion the power excursion is determined mainly by the fuel temperature feedback alone. On the other hand, a much slower insertion by natural circulation will allow significant negative feedback from the coolant density as well.

Two reactivity values of potential core disturbances are of particular interest, the criticality limit ($k_{eff}=1$) and the severe reactivity initiated accident (RIA) limit k_{RIA}, beyond which the hottest fuel rods may already disintegrate. From basic reasoning applying the Fuchs-Hansen model for rapid power excursions this limit is estimated to be at least $k_{RIA}=1.025$ for fast reactivity insertions. This result is derived in Appendix 1. The effects of power peaking are included in this estimate. With slow reactivity insertion rates the consequences become milder. The static k_{RIA} serves as a conservative screening limit for severe reactivity accidents in VVERs. The core design code

HEXBU-3D [7] was used to study the static reactivity effects of diluted slugs of different geometries in the core. Different reactor states, either critical or shut down with control rods were analyzed. Results are shown in Figures 3, 4, and 5.

In most cases the reactor is shut down with control rods at the time the diluted slug enters into the core. From the probabilistic point of view the frequency of boron dilution accidents should be designed to be very low. Since the probability of such an event with a stuck control rod must be several orders of magnitude even lower, it is reasonable to consider severe reactivity accidents mainly with all control rods inserted after a reactor scram.

4. DYNAMIC ANALYSES OF CORE RESPONSE

The estimated severe RIA limit of reactivity was confirmed with the reactor dynamics code HEXTRAN [8]. HEXTRAN is a three-dimensional reactor core dynamics code for coupled neutron kinetic and thermal-hydraulic calculation of VVER type reactors. Hot channel analysis capability including local fuel and cladding temperatures and cladding oxidation is also provided. HEXTRAN is dynamically coupled with the code SMABRE [9], which is a thermal-hydraulic model of the primary and secondary circuits. All six loops are modeled separately and the reactor vessel is modeled in six interconnected sectors. The coupled code was used in the analyses.

It is well known that numerical diffusion tends to destroy the sharp boron fronts in codes which use the standard solution methods. This problem was avoided by simulating the dilution front directly to the core inlet of HEXTRAN as external time-dependent input. The timing, duration and volumes were evaluated from a precalculation with SMABRE alone. The flow velocity and assumed mixing are of great importance.

The results for three external dilution scenarios in hot reactor conditions are summarized below as examples. In these scenarios the coolant flow rate varies considerably at the time of slug injection into the core.

4.1 Incorrect start-up of an isolated loop (Case A)

In this case the reactor is initially at 1 % power. Five reactor coolant pumps are in operation and one loop is isolated by a closed main gate valve in the cold leg. A diluted slug of 3 m^3 is assumed in the hot leg. Incorrectly, the last reactor coolant pump is started first and then the main gate valve is opened. It was assumed that the flow from the accelerating loop enters its wn 60° sector, where it mixes with water from the other loops until full flow rate fc the starting loop is reached. In the base case no credit was given for mixing of the dilution front in passing through the steam generator. The original dilution in the hot leg was assumed to be -913 ppm, which is reduced to -730 ppm in the core due to a lower flow rate in the starting loop. This disturbance has a static reactivity potential k_{eff}=1.030 when the slug fills a 60° of the core. It was selected in order to reach the limiting fuel pellet average enthalpy in the hot fuel rod. The core power distribution is strongly peaked in the disturbed sector.

The power production in the core is shown in Figure 6. The rise to power begins with a prompt supercritical power excursion. The diluted slug passes through the core in about 2.5 seconds. As undiluted water enters the core again the power is thereby rapidly reduced. A final reduction in power occurs as the control rods enter the core due to reactor scram. The total energy production during the event is about 9800 MWs. About 2000 MWs is produced during the initial prompt supercritical excursion.

The calculated peak pressure in the primary circuit is 135 bar. This is just below the opening pressure of the first safety valve at 137 bar.

Maximum temperatures in the hot fuel rod are shown in Figure 7 as a function of time. The maximum fuel pellet average enthalpy is 960 J/g (229 cal/g). A fuel certerline temperature exceeding 2800 °C is not physical and represents fuel enthalpy beyond the melting point of UO_2. The rapid rise in fuel cladding temperature is due to DNB. The maximum calculated cladding temperature is 1660 °C, but the maximum local cladding oxidation is only 5 %. Fuel cladding failures (leakages) induced by the high enthalpy of the fuel and by DNB can be expected.

This example is essentially at the limit of 963 J/g (230 cal/g) for fuel pellet average enthalpy acceptable in Finland during a reactivity accident, disregarding high burnup effects. Beyond this limit severe fuel damage is assumed. It is typical for fast reactivity accidents that the LOCA limit 1200 °C for fuel cladding temperature is exceeded for a short time, but the cladding oxidation is well below the limit of 17 %. Some fuel centerline melting occurs, but due to its central hole the VVER fuel pellet can easily accommodate some volume expansion of the melting fuel. In this example about 20 % of the hot pellet volume is estimated to melt. Ultimately, volume expansion during fuel melting is the driving force which can produce a rapid dispersion of the fuel.

This example also demonstrates that the static reactivity potential k_{eff}=1.025 of the diluted slug is a reasonably conservative screening limit for severe reactivity accidents in VVER-440 reactors.

4.2 Restart of the first reactor coolant pump (Case B)

In this case the reactor is shut down in the hot state with all control rods inserted. All the reactor coolant pumps have stopped during start-up dilution. It is assumed that diluted water has been accumulated in one loop (16 m^3) due to continued dilution and stagnant coolant conditions in the primary circuit. When the first reactor coolant pump is restarted 12 m^3 of diluted water is directed towards the core and enters the whole core at the full flow rate produced by one pump with all loops open. In the base case the boron disturbance of the slug at core inlet was assumed to be -1286 ppm with respect to the hot critical boron concentration. A core inlet temperature disturbance

-30 °C was also assumed. This disturbance has a static reactivity potential k_{eff}= 1.025 when the slug fills the entire core.

The power production in the core is shown in Figure 8. There is again an initial prompt supercritical power excursion. After the excursion the power remains around 600 MW until undiluted water re-enters the core thereby rapidly reducing the power. The diluted slug passes through the core in about 8 seconds. The total energy production during the event is about 4800 MWs. About 1300 MWs is produced during the initial prompt supercritical excursion.

The calculated peak pressure in the primary circuit is only 131 bar, although the whole core takes equally part in the power production.

Maximum temperatures in the hot fuel rod are shown in Figure 9 as a function of time. The maximum fuel pellet average enthalpy is only 390 J/g (93 cal/g). The hot fuel rod experiences DNB with a small delay after the initial power excursion. This raises the cladding temperature up to 1100 °C during the rather long duration of the disturbance. The maximum local cladding oxidation remains around 0.5 %. Thus only cladding failures induced by DNB are to be expected.

4.3 Restart of natural circulation (Case C)

In this case the reactor is also shut down in the hot state with all control rods inserted. All the reactor coolant pumps have stopped during start-up dilution. It is assumed that diluted water has been accumulated in the primary circuit due to continued dilution and stagnant coolant conditions. Later recovery of natural circulation is then assumed to transport the diluted water into the whole core with the flow rate of natural circulation corresponding to 0.5 % decay heat power in the core and all six loops open. In the base case the boron disturbance of the diluted slug at core inlet was assumed to be -1409 ppm with respect to the hot critical boron concentration. This disturbance has a static reactivity potential k_{eff}=1.025 when the slug fills the entire core. The diluted slug was described as endless (semi-infinite) in order to reach the stationary state of the disturbed core in one computer run.

The power production in the core is shown in Figure 10. There is again an initial, barely prompt supercritical power excursion that essentially brings the very low neutron flux up to a level were nuclear feedback effects can take over. About 1000 MWs of energy is produced during the first few seconds of the initial excursion. After some oscillations coupled with steam production in the core the power stabilizes at about 250 MW.

In spite of the increasing flow rate of natural circulation there is a large time delay in transporting heat into the steam generators. Therefore, there is a steady expansion of the primary coolant and an increase of the primary circuit pressure until the safety valve opens for the first time when about 30 m^3 of diluted water has entered into the core. If the size of the slug is limited to 15 m^3 then the maximum pressure will be approximately 130 bar.

Maximum temperatures in the hot fuel rod are shown in Figure 11 as a function of time. In this case DNB does not occur in spite of the low flow rates during natural circulation. The peak fuel pellet average enthalpy is only 155 J/g (37 cal/g) and the peak cladding temperature is only 340 °C.

The above cases demonstrate how a slower rate of reactivity insertion into the core tends to make the incident milder. If the reactivity disturbance is further increased during natural circulation, DNB and correspondingly the fuel cladding temperature and oxidation become the limiting parameters. The peak fuel pellet enthalpy seems to be of lesser concern in the case of natural circulation.

4.4 Cold reactor states

Experience with analyses of diluted slugs in the cold state indicates that there are several phenomena which make the situation worse than in the hot state. First, the reactivity potential of diluted slugs is higher for the cold reactor. Second, at low pressures the volume expansion of water during boiling is much larger. This tends to create calculational problems and evidently also real problems with the temporary drying of hot channels thus enhancing DNB. Third, at low pressure heat transfer during film boiling is less effective and tends to raise the cladding temperatures even at low heat flux densities. The main threats in the cold state seem to be cold overpressurization of the primary circuit and fuel cladding melting.

5. PREVENTIVE MEASURES AND PSA

All systems of VVER-440 reactors have not been designed with the aim of minimizing the potential for formation of local diluted slugs. Details may, however, vary from one plant to the other. New automation and operating procedures have been implemented at Loviisa NPP to upgrade the protection against the inadvertent formation and injection of diluted slugs of water into the core. For hot reactor operating conditions these actions have the following purpose:

* to interrupt dilution operations promptly
* to ensure that the boron concentration in the pipelines of the make-up water system is not too low
* to ensure that all disturbances are removed from a loop whenever this is possible before starting the reactor coolant pump.

Dilution is automatically interrupted by several diverse automatic actions when one or more reactor coolant pumps stop. The suction from the degasifier for pure condensate is closed and shifted to the degasifier for normal make-up water. High capacity make-up pumps are stopped, unless they are injecting boron solution. In addition, when more than three reactor coolant pumps stop, high capacity boron supply to the suction of the make-up pumps is started.

The mixing of pure condensate with the recirculation flow in the coolant purification system is ensured by limiting the flow rate of pure condensate and by monitoring the mixing conditions with the process computer system. This is to ensure that the boron concentration in the pipelines of the make-up water system is not lower than what is acceptable according to the reactivity accident analyses. In 1994 a separate borated dilution water tank was installed, from which dilution water of specified minimum boron concentration is taken to the former degasifier of pure condensate to ensure that dilution limits are not exceeded in any case. The boron concentration in this tank is kept at most 1200 ppm below the hot zero power critical boron concentration of the reactor. The absolute concentration decreases with cycle burnup and becomes zero at about mid-cycle.

The flushing of closed loops by reverse flow is enforced by new automation that requires the main gate valves to be fully open before a pump can be started. The automation includes an intermediate partial opening of the valve for 400 seconds to limit the initial flushing rate.

When all the reactor coolant pumps have stopped, the first pump to be restarted shall be selected from a loop with no make-up water injection.

Performed PSA studies indicate that these actions have reduced the probability of a severe reactivity accident due to local boron dilution during hot operating conditions of the reactor to less than 10^{-6} per year. In practice this figure reduces to the order of 10^{-7} per year, when credit is taken for modest mixing assumptions together with three-dimensional analysis of core behaviour for dilutions limited to 1200 ppm below the hot zero power critical boron concentration.

6. SUMMARY AND CONCLUSIONS

In this study for Loviisa NPP the core response has been analyzed assuming that a slug of diluted water enters the core by either forced flow or by natural circulation. Both hot and cold reactor conditions have been studied. Conservative assumptions on coolant mixing have been used and the slug is assumed to enter either into one sector of the core or into the whole core as a layer.

Three-dimensional dynamic calculations of core response with the HEXTRAN code were performed to confirm the allowable minimum boron concentration for the dilution water. It was found that the severe RIA limit for the static multiplication factor of the disturbed core is $k_{RIA} \geq 1.025$ in hot coolant conditions, corresponding to a peak fuel pellet average enthalpy limit 963 J/g (230 cal/g). The lower limit applies to fast reactivity insertions with essentially fuel temperature feedback only. It is clearly beyond mere prompt criticality. At reduced coolant flow rates there is additional margin due to the slower reactivity insertion rate in itself and due to reactivity feedback from coolant density. In the case of natural circulation DNB and correspondingly fuel cladding temperature and oxidation become the limiting parameters.

The results from analyzing the core response in potential boron dilution accidents during reactor startup and power operation lead to the following general conclusions for hot coolant conditions. When the boron concentration in the makeup water injected into the primary circuit is at most 1200 ppm below the hot critical concentration, there is practically no risk of a reactivity accident with severe fuel damage. This conclusion can be reached using rather conservative assumptions on mixing in the steam generators and in the reactor. When all control rods are inserted into the core it appears to withstand without severe fuel damage even the injection of small slugs of pure water less than 12 m^3 in size by natural circulation at beginning of cycle. Plant modifications have been implemented to ensure that the makeup water during dilution operations has a safe boron concentration and to reduce the probability of external local boron dilutions.

The results from analyzing the core response in cold coolant conditions and with all control rods inserted indicate that there is only a relatively narrow band in the boron concentration of a diluted slug between reaching criticality and reaching the onset of severe damage. The major dangers appear to be the cold pressurization of the closed primary circuit and the melting of fuel rod cladding in connection with a heat transfer crisis. When the static reactivity potential of a diluted slug is less than 2 % and its size is less than 12 m^3, severe damage may still be avoided. In cold shutdown conditions there are essentially two defenses against severe reactivity accidents: administrative procedures to exclude diluted water from entering the primary circuit and coolant mixing phenomena as an inherent protection against small amounts of diluted water.

The degree of inherent protection provided by coolant mixing phenomena in the steam generators, loop piping and reactor vessel requires further study. These studies are being performed by using the hydrodynamics code PHOENICS.

REFERENCES

[1] D.J.Diamond, C.J.Hsu, R.Fiztpatrick: Reactivity Accidents, A reassessment of the Design-Basis Events. Brookhaven National Laboratory, NUREG/CR-5368 (BNL-NUREG-52198), Jan. 1990.

[2] S.Jacobson: Risk Evaluation of Local Dilution Transients in a Pressurized Water Reactor. Lindköping University, Studies in Science and Technol., Dissertations No. 275, 1992.

[3] P.Siltanen, M.Antila: Extended Protection against Reactivity Accidents Caused by Slugs of Diluted Water in the Loviisa Reactors. Second Symposium of AER. Paks, Hungary, 21-26 Sept. 1992.

[4] P.Siltanen, M.Antila, R.Kyrki-Rajamäki: Results of Core Response in Reactivity Accidents due to Local Boron Dilution. Fourth Symposium of AER, Sozopol, Bulgaria, 10-15 Oct. 1994.

[5] J.Hyvärinen: The Inherent Boron Dilution Mechanism in Pressurized Water Reactors. Nuclear Engineering and Design 145 (1993) 227-240.

[6] P.Gango: Numerical Boron Mixing Studies for Loviisa Nuclear Power Plant. Seventh Int. Mtg. on Nuclear Reactor Thermal-Hydraulics (NURETH-7), Saratoga Springs, NY, USA, Sept. 1995.

[7] E.Kaloinen, P.Siltanen, R.Teräsvirta: Two-group Nodal Calculations in Hexagonal Fuel Assembly Geometry. OECD/NEA Specialists' Mtg. on the Calculation of 3-Dimensional Rating Distributions in Operating Reactors, Paris, Nov. 26-28, 1979, pp. 111-128.

[8] R.Kyrki-Rajamäki: Three-Dimensional Reactor Dynamics Code for VVER type Nuclear Reactors. Technical Research Centre of Finland, Espoo, 1995. 51p. + App. 80 p. (VTT Publications 246).

[9] J.Miettinen: Development and Assessment of the SBLOCA Code SMABRE. Proceedings of the Specialists Meeting on Small Break LOCA Analyses in LWRs, Pisa, Italy, 1985.

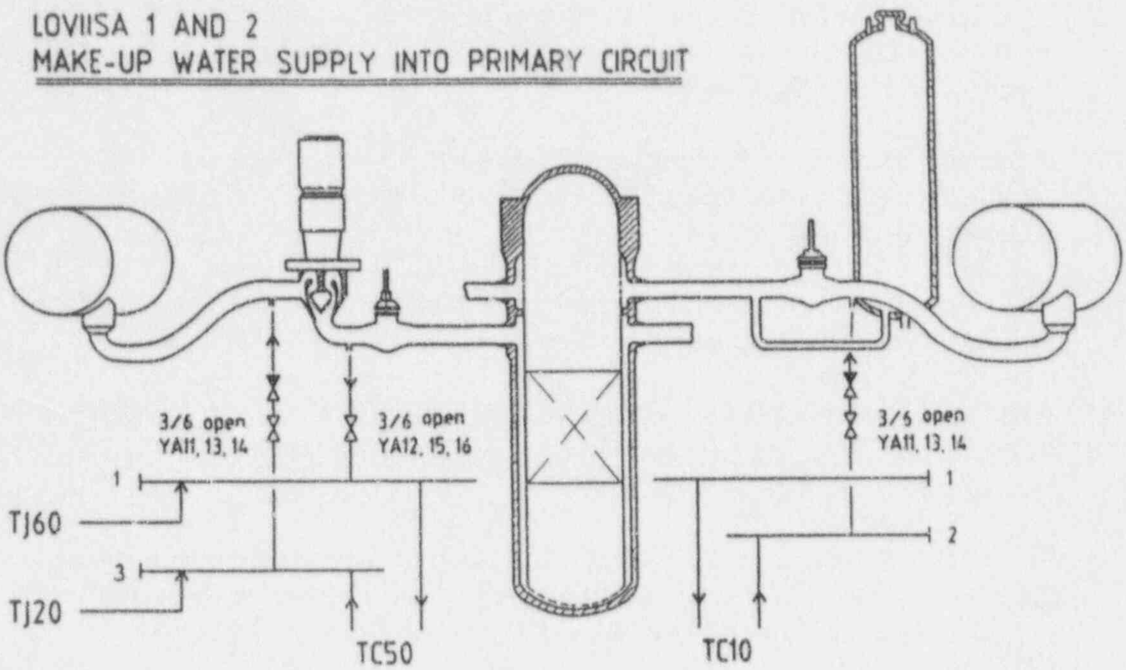


Figure 1. Layout of the Loviisa VVER-440 primary circuit and injection of make-up water (TC) into the six loops (YA11 to 16).

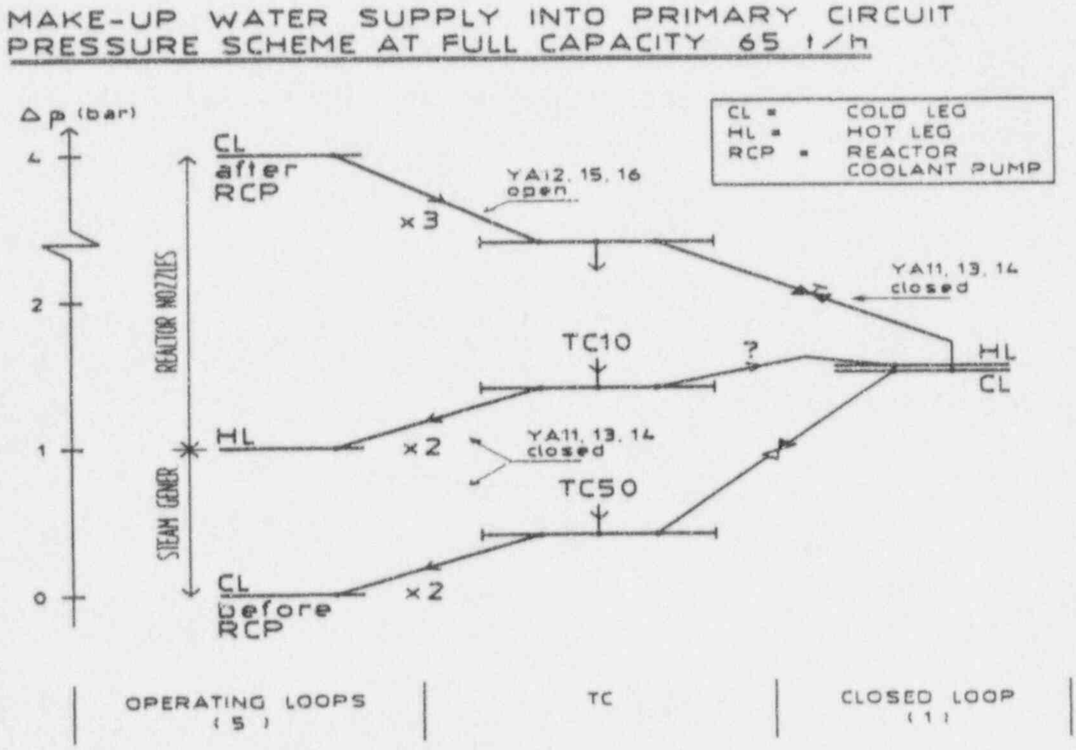


Figure 2. Pressure scheme for the primary circuit injection points and make-up water supply lines at full injection capacity 65 t/h.

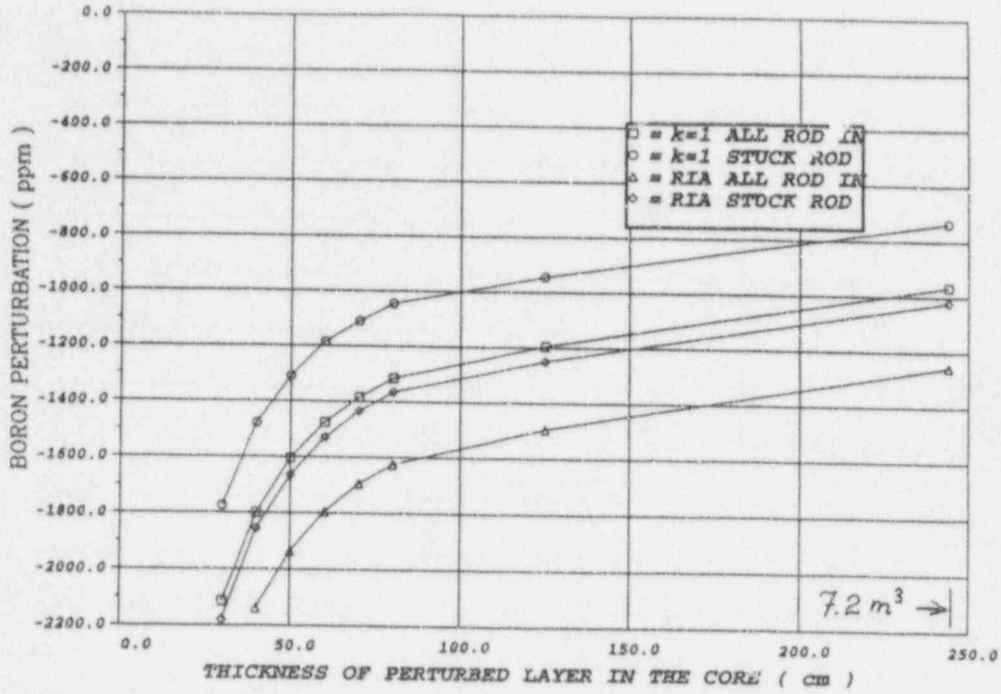


Figure 3. Limits of recriticality (k = 1) and severe RIA potential (k = 1.025) for a boron diluted layer in the core at BOC after scram of hot critical reactor.

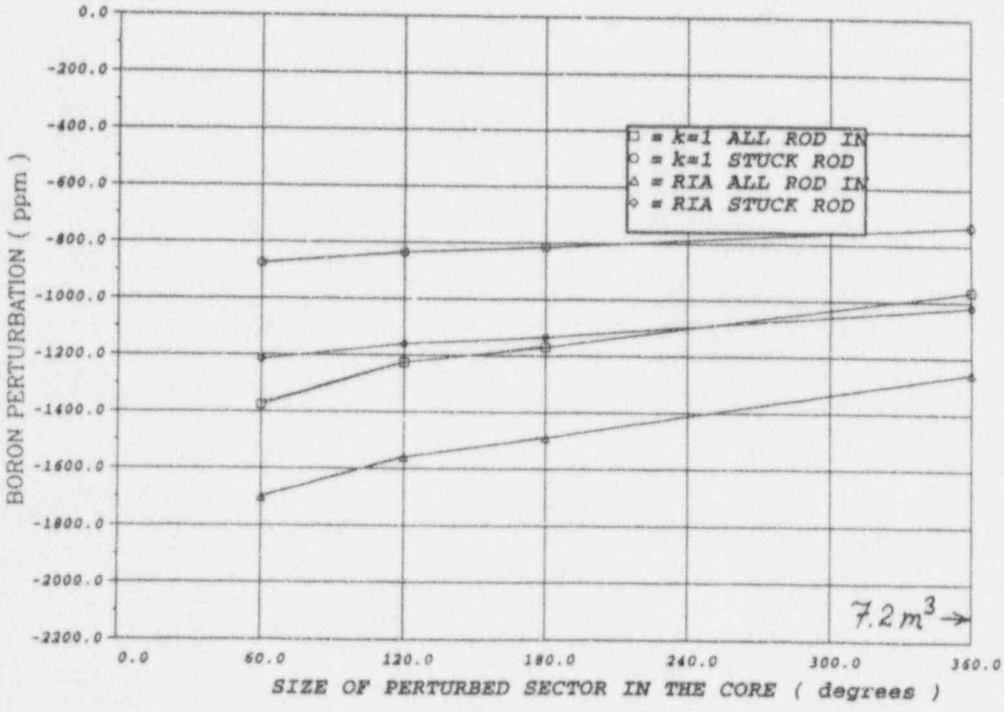


Figure 4. Limits of recriticality (k = 1) and severe RIA potential (k = 1.025) for a boron diluted sector in the core at BOC after scram of hot critical reactor.

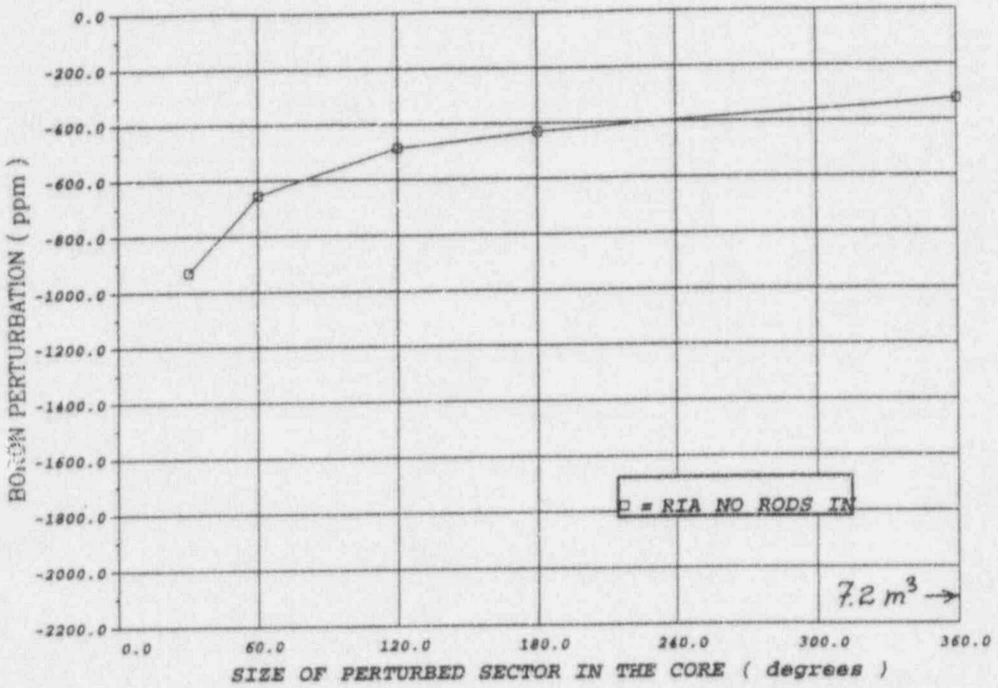


Figure 5. Limit of severe RIA potential (k = 1.025) for a boron diluted sector in the core at BOC in a hot critical reactor.

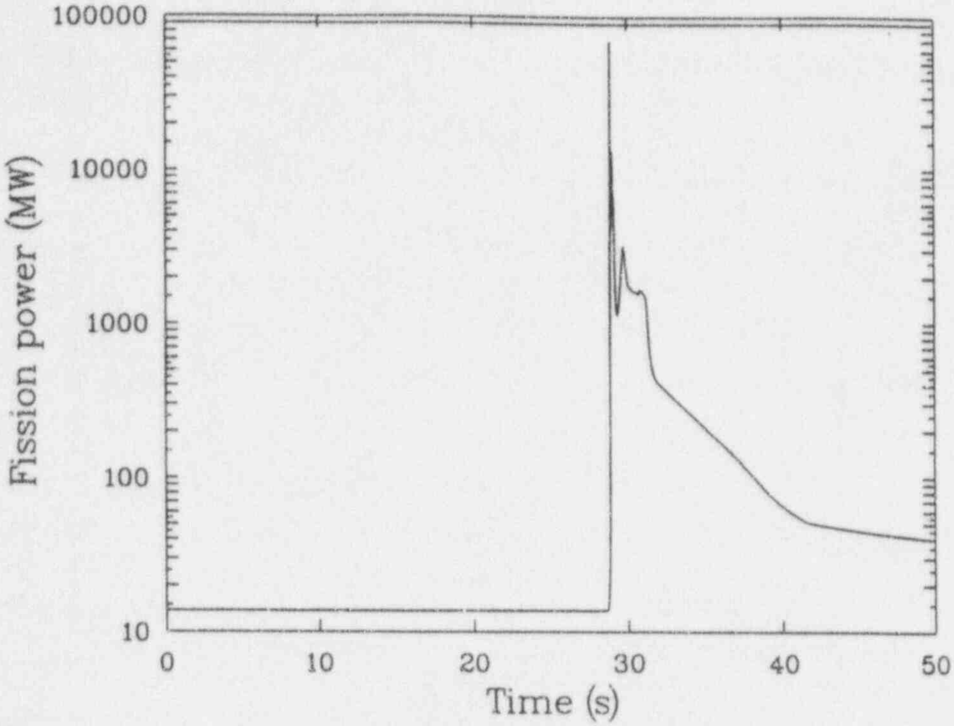


Figure 6. Core fission power during incorrect start-up of an isolated loop (Case A).

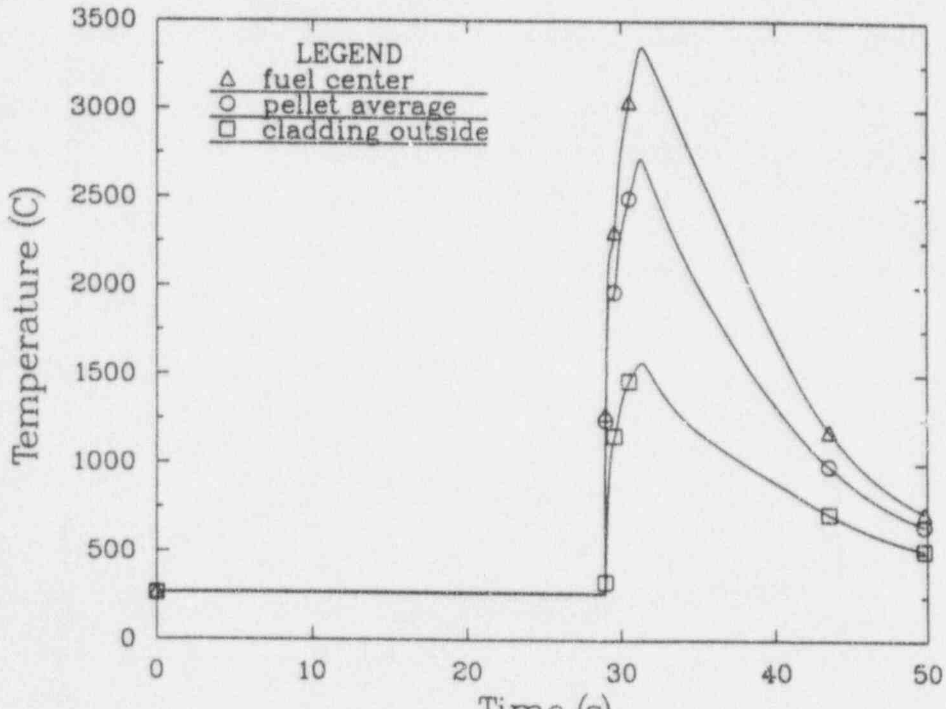


Figure 7. Maximal temperatures in hot fuel rod during incorrect start-up of an isolated loop (Case A).

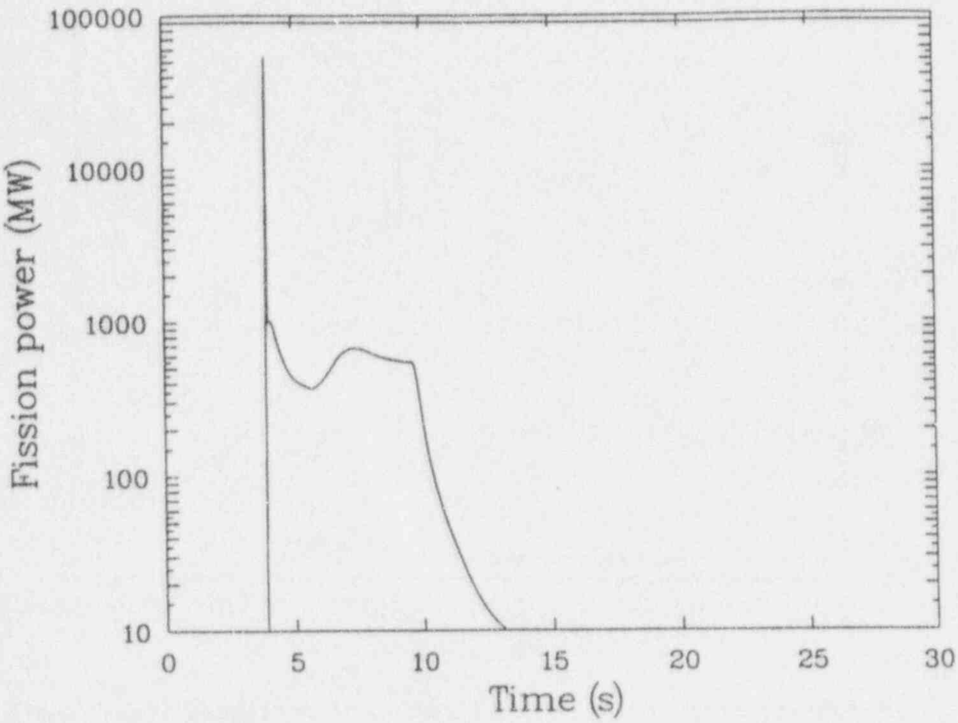


Figure 8. Core fission power during restart of the first reactor coolant pump (Case B).

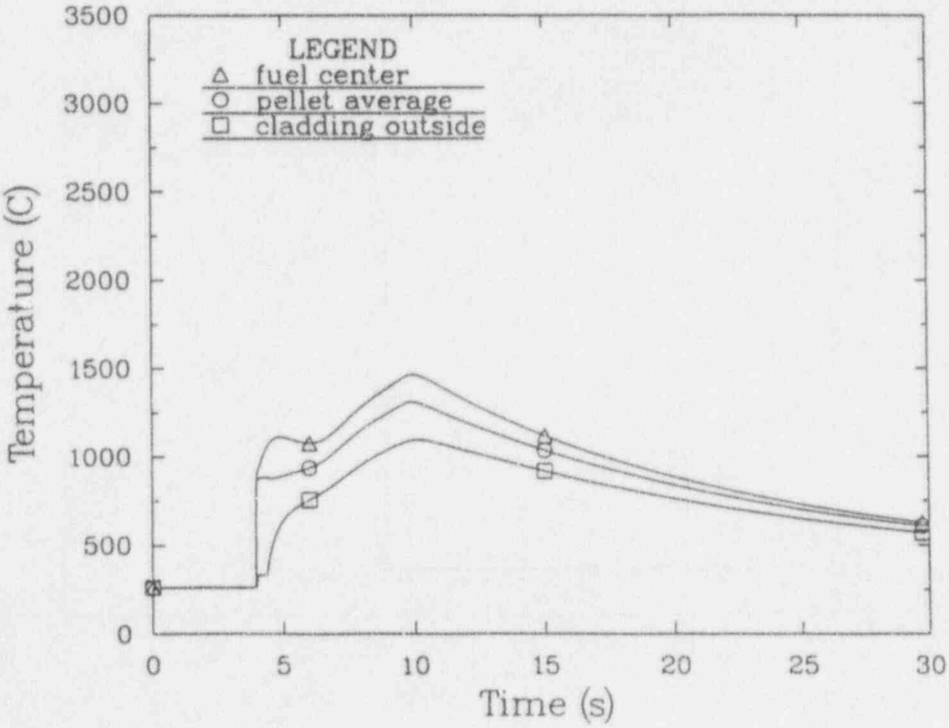


Figure 9. Maximal temperatures in hot fuel rod during restart of the first reactor coolant pump (Case B).

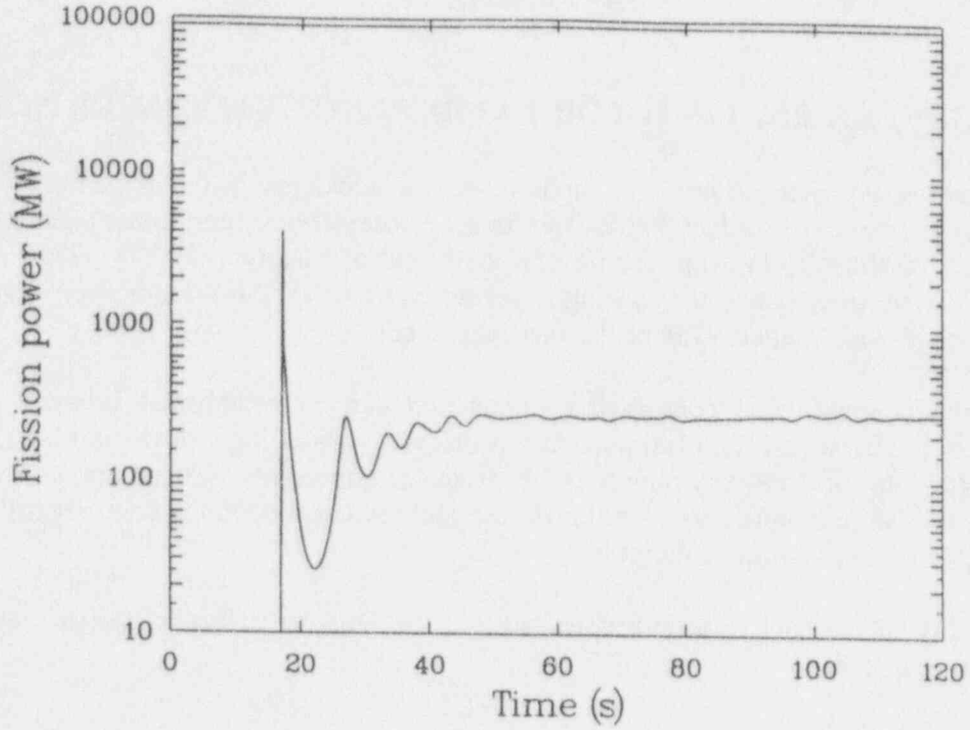


Figure 10. Core fission power during restart of natural circulation (Case C).

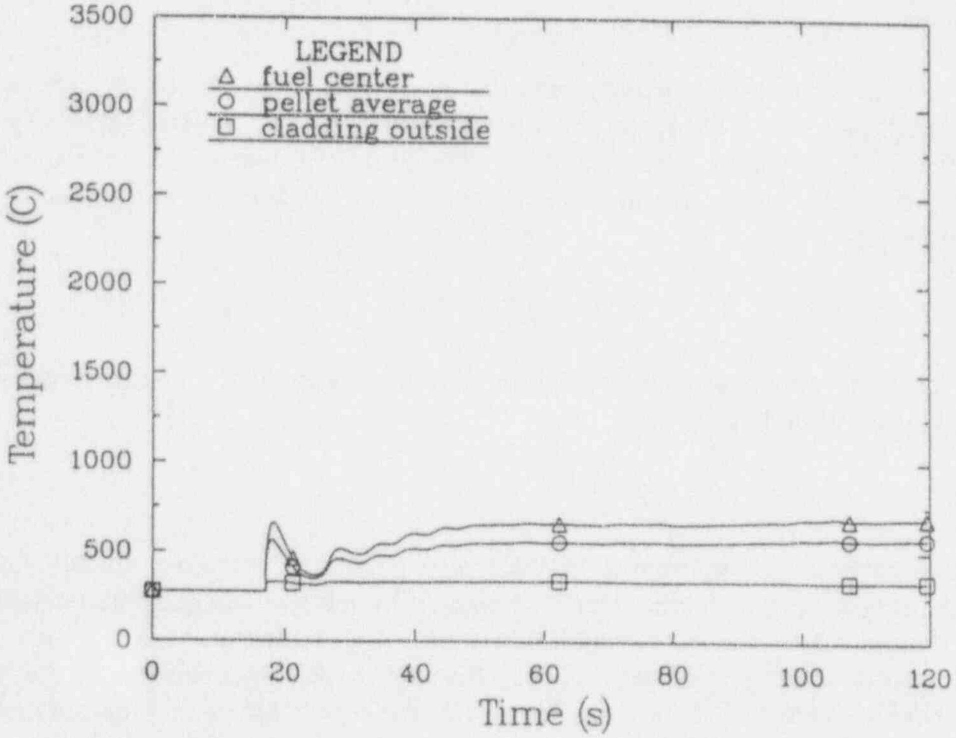


Figure 11. Maximal temperatures in hot fuel rod during restart of natural circulation (Case C).

APPENDIX 1

THE SEVERE RIA LIMIT FOR RAPID REACTIVITY INSERTION

The main acceptance criterion in rapid reactivity accidents is the maximum fuel enthalpy averaged over a fuel pellet. The limiting acceptable value for this quantity is 963 J/g (230 cal/g) in Finland. This corresponds approximately to 3000 K (2730 °C), which is somewhat below the melting temperature of UO_2. Above this limit severe fuel damage, i.e. disintegration of the fuel may occur.

A relatively simple and conservative connection can be established between the reactivity potential and the maximum fuel enthalpy by assuming that the diluted slug is injected into the core very rapidly. This produces a promptly supercritical pulse of energy with an approximately fixed maximum fuel temperature in the core, regardless of the power distribution of the pulse.

The ability of the fuel to absorb reactivity via the Doppler effect is approximately given by the relation

$$\Delta\rho = C\ \Delta\sqrt{T_F}\ .$$

The coefficient C can be linked with the normal fuel temperature coefficient of reactivity. For a typical temperature coefficient -2.7 pcm/K at 900 K we obtain C = -160 pcm/√K.

Let us assume a uniform reactivity perturbation in the core. When the hot pellets reach a temperature of 3000 K the average enthalpy rise of all pellets is typically half of the hot pellets, taking some credit for the prompt influence of the rising fuel temperature on the power distribution. Accordingly, the average fuel enthalpy and temperature are

$$i_F = 530\ \text{J/g}\ , \quad T_F = 2000\ \text{K}\ (1730\ ^\circ\text{C})\ , \quad \Delta\sqrt{T_F} = 22\ \sqrt{\text{K}}\ .$$

The corresponding reactivity change and thereby the ability of the fuel to compensate reactivity at the RIA limit is

$$\Delta\rho = -3500\ \text{pcm}\ \ (-3.5\ \%)\ .$$

This same estimate is approximately valid also when the reactivity perturbation is limited to only part of the core. This is explained by the fact that the reactivity of the core is dominated by the perturbation and by the fuel temperature at the local flux peak. In a way, the region of the flux peak forms a smaller, independent core with an internal peaking factor of about two. The rest of the core behaves as a multiplying but still subcritical reflector. Ultimately this concept is verified by explicit three-dimensional simulations of such power excursions.

The need for the ability of the fuel to compensate reactivity is largest for a prompt supercritical energy pulse. In this case the amount of reactivity exceeding the prompt criticality limit has to be compensated two-fold according to the Fuchs-Hansen model [1]. Single compensation merely stops the prompt growth of the neutron flux when the pulse has delivered half of its energy. Accordingly, at the RIA limit of the fuel enthalpy the reactivity insertion is limited to such a value that

$$2\,(\rho - \beta_{eff}) \;=\; 3500 \text{ pcm} .$$

With β_{eff} = 600...650 pcm at beginning of cycle, the static reactivity in a fast reactivity insertion should be limited to the value

$$\rho \;\leq\; 2400 \text{ pcm} \qquad \text{or} \qquad k_{eff} \;\leq\; 1.025 .$$

This result is in line with three-dimensional dynamic analyses of boron perturbations entering a 60° sector of the core at essentially full flow rate. This static reactivity limit can serve as a conservative screening limit for severe reactivity accidents in VVER-440 reactors.

In practice, the reactivity insertion rate of a diluted slug is usually slow enough to provide additional capability for compensating the inserted reactivity by the fuel temperature and by the coolant density. As the diluted slug leaves the core the reactor becomes borated again.

It is important to understand properly the significance of local power peaking in the core, such as that produced by a localized dilution in the core or due to a stuck control rod. The reactivity of the core is dominated by the events in the region of the flux peak. Therefore, a given potential supercriticality will approximately lead to the same maximum fuel enthalpy. For a given maximum fuel enthalpy the total energy produced in the core will depend on the power distribution. The higher the power peaking is the smaller will be the total energy release and the consequences related to it, such as the pressure peak in the primary circuit. Low power peaking will maximize the total energy output.

References

[1] G.I.Bell, S.Glasstone: Nuclear Reactor Theory. Van Nostrand Reinhold Co., New York, 1970. pp. 517-520.

COMPARATIVE STUDY OF A BORON DILUTION SCENARIO IN VVER REACTORS

K. Ivanov*, U. Grundmann, S. Mittag, U. Rohde

Research Center Rossendorf Inc., Institute for Safety Research
P.O. Box 510 119, D-01314 Dresden

ABSTRACT

Subsequent studies have identified many scenarios which can lead to reactivity excursions due to boron dilution. The comparative study, presented in this paper, deals with the so-called "restart of the first reactor coolant pump" scenario and its reactor-dynamic consequences for both VVER reactor types -VVER-440 and VVER-1000.

The transient simulations have been performed using the three-dimensional core dynamics code DYN3D. The DYN3D modeling features, including recent developments, as well as the cross-section generation methodology, involved in these calculations, are described. The analyzed accident scenario is outlined together with the assumptions made. The results of core response in this boron dilution accident for both VVER reactors have been compared within the ranges, determined by the two reactivity values of interest: the criticality limit and the reactivity initiated accident (RIA) limit.

1. Introduction

Subsequent studies have identified many scenarios for PWR's which can lead to reactivity excursions due to boron dilution. These events can be caused by different mechanisms which are usually classified in two categories: external and inherent boron dilution mechanisms. The typical external dilution scenarios for the VVER-440 reactors, as well as the potential consequences in the core at hot reactor conditions have been recognized and studied using the Loviisa Nuclear Power Plant (NPP) as a reference design[1,2].

Some studies have been accomplished at Research Center Rossendorf (RCR) to analyze reactivity accidents during the incorrect start-up procedure of an isolated loop in a VVER-440 reactor[3,4]. These transients are characterized by significant space dependent effects due to the asymmetric boron perturbation. Using the core dynamics code DYN3D[5] three-dimensional (3-D) full-core simulations have been performed to access the potential risk of fuel damage. The effect of the applied mixing model has been investigated also. Further the fast running code SITAP, designed for the primary system simulation, have been used to provide DYN3D with more realistic time-dependent boundary conditions in modeling boron dilution transients[6].

---

*On leave from the Institute of Nuclear Research and Nuclear Energy, Sofia.

2. Three-Dimensional VVER Core Simulation

The DYN3D code, developed at RCR, was used to perform this comparative analysis. DYN3D is designed for a 3-D transient simulation of reactor cores with hexagonal geometry. It includes the modeling of 3-D core kinetics, two-phase flow and heat transfer, fuel rod behavior and coolant mixing in the downcomer and lower plenum. The time-dependent conditions at core boundaries have to be given as input.

The spatial neutronic model is based on a nodal expansion method for solving the two-group neutron diffusion equation in hexagonal-Z geometry. The partial current formulation is utilized. In this method the space dependence of neutron fluxes inside the nodes is separated in the hexagonal plane and in the axial direction. An expansion with Bessel functions is used to approximate 2-D flux in radial plane while the standard polynomial expansion up to the fourth order is applied in axial direction. The Chebyshev technique is used to accelerate convergence of the outer iteration cycle. The temporal model is based on an implicit finite-difference scheme with implemented exponential transformation technique. The DYN3D neutronics was verified on 2-D and 3-D steady-state and kinetics benchmarks, as well as against some measured data and results from kinetic experiments.

A burnup version of DYN3D has been recently developed. This provides a capability to independently calculate 3-D exposure distributions at different points in the cycle depletion, taking into account the history effects. In this way, the initial steady-state conditions and the following transient simulation are modeling in a consistent manner using the same core neutronics. The burnup version was tested on a VVER-440 benchmark problem for the Paks NPP[7]. The initial exposure distribution was given at the end of the third cycle of Paks unit 2. Starting from this point, the depletion had to be calculated through cycles 4 to 7 using the corresponding shuffle schemes and operational histories provided by the utility. The burnup distribution obtained by DYN3D at the end of cycle 7 was compared to reference results provided by Paks NPP[7] (Fig. A). The maximum relative deviations in burnup are less than 5 %, and absolute differences are not greater than 0.5 MWd/kg. The largest relative deviations are observed at the core boundaries.

The thermal-hydraulics of DYN3D comprises a two-phase flow model and a fuel rod model. The fuel elements are simulated by separate coolant channels. Hot channel analysis capability is also available. Several safety parameters as local fuel and cladding temperatures, cladding oxidation, DNBR and fuel enthalpy are evaluated. Special mixing models for the downcomer and the lower plenum of VVER reactors have been implemented in DYN3D. The thermal-hydraulic model of DYN3D was validated by comparison with benchmarks, results of other codes and experiments. The following perturbations can be simulated during a transient calculation: movements of single control rods or control-rod groups; variation of core coolant inlet temperature; variation of boron concentration; changes of core pressure drop or total mass flow rate; and changes of pressure. The DYN3D control-rod model has two implemented options: one is using the standard geometrical weighing approach while the other is utilizing a more sophisticated flux-volume weighing technique.

The macroscopic cross sections and the parametrization coefficients are included in the data set of neutron kinetics. The data are written on a separate file, which is read by the code. In VVER-440 calculations the two-group data are obtained from the MAGRU library[8], which was generated by KAB Berlin using the code NESSEL-4 and the nuclear data from ENDF/B4. The group data Σ (macroscopic cross sections and diffusion coefficients) depend on burnup $\boldsymbol{A}$, fuel temperature $\boldsymbol{T_F}$, moderator temperature $\boldsymbol{T_M}$, moderator density ρ_M, and boron acid concentration $\boldsymbol{C_B}$. They are calculated in DYN3D for each node in the following way:

$$\Sigma(A, T_F, R_M, C_B) = \Sigma^0(A) \times K_{TF}(A) \times K_{RM}(A) \times K_{CB}(A)$$

$$K_{TF}(A) = 1 + \alpha_{TF}(A) \left(\sqrt{T_F} - \sqrt{T_M}\right)$$

$$\alpha_{TF}(A) = a_0^{TF} + a_1^{TF}A + a_2^{TF}A^2$$

$$K_{RM}(A) = 1 + \alpha_{RM}(A) \left(\rho_M - \rho_M^0\right) + \beta_{RM}(A) \left(\rho_M - \rho_M^0\right)^2$$

$$\alpha_{RM}(A) = a_0^{RM} + a_1^{RM}A + a_2^{RM}A^2$$

$$\beta_{RM}(A) = b_0^{RM} + b_1^{RM}A + b_2^{RM}A^2$$

$$K_{CB}(A) = 1 + \alpha_{CB}(A) \left(C_B\rho_M - C_B^0\rho_M^0\right) + \beta_{CB}(A) \left(C_B\rho_M - C_B^0\rho_M^0\right)^2$$

$$\alpha_{CB}(A) = a_0^{CB} + a_1^{CB}A + a_2^{CB}A^2$$

$$\beta_{CB}(A) = b_0^{CB} + b_1^{CB}A + b_2^{CB}A^2$$

The reference group data Σ^0 are interpolated for the actual burnup value from burnup sampling points given in the MAGRU library, which also contains the coefficients a_0, a_1, a_2 and b_0, b_1, b_2.

A similar llibrary, generated by Energoproject Prague[9] with the help of the Russian code KASSETA, is used in the case of VVER-1000.

3. Comparative Calculations of a Boron Dilution Scenario

The purpose of this study is to compare the results of core response in a reactivity transient following the same boron dilution scenario in both VVER reactor types. VVER-1000/V320 and VVER-440/230 are used as reference designs[10,11]. In order to simulate compatible conservative initial conditions for both reactors, the beginnig of cycle (BOC) state for a fresh core has been selected. These are typical high leakage designs. No fission product poisoning is assumed.

The "restart of the first reactor coolant pump" scenario, described in Ref. 1 and 2, is followed. The reactor is assumed to be close to criticality during the start-up dilution or to be operating at power when all the reactor coolant pumps stop. Reactor scram is then activated and the reactor becomes subcritical with all the control rods inserted. It is assumed that diluted water has been accumulated in one loop due to continued dilution and stagnant coolant conditions. If the pump in this loop is the first one to be started a slug of diluted water can be transported to the core. The slug will enter the core as a layer spreading over the core cross section at the flow velocity of one coolant pump. In a conservative approach the case of whole core disturbance is analyzed. Further, following the above mentioned references, the values of core disturbance corresponding to the static criticality limit (k_{eff}=1) and the RIA limit k_{RIA} were calculated. Only dilution perturbation is assumed and the corresponding dilution values for both reactors were evaluated. Two cases have been analyzed depending on the assymptions made for the initial steady-state conditions. The first one is a subcritical state with all the control-rod groups inserted according to the original scenario described above. The second one is assuming that during the scram the most reactive control rod was stuck out. The results obtained for the first case are presented in this paper. Calculations of the second case are now being performed. The loop volumes, the flow velocities of one coolant pump, and the reacor core geometrical characteristics are different for VVER-440 and VVER-1000 and it results in different transient evolutions.

The time behaviors of reactor power are given in Fig. 1 (VVER-1000) and Fig.4 (VVER-440). The corresponding reactivity curves are displayed in Figs. 2 and 5. In the case of VVER-1000 reactivity insertion caused by the boron dilution is compensated first of all by Doppler feedback effect. After compensation of power peak the reactor behavior is determined by thermal-hydraulics. Boiling occurs in parts of the core and moderator density feedback is significant. In the VVER-440 core the consequences are much more severe. The reactivity induced by the dilution layer produces essentially a prompt pulse. The power peak is very high. Figs. 3 and 6 show the maximal fuel and cladding temperatures as a function of time for both reactors. For VVER-1000 the maximum fuel pellet enthalpy is 320 J/gUO_2. Maximum cladding temperature is only 350° C. The VVER-440 calculation is interrupted after 2 s because the limits of the model application are reached. The reasons for obtaining such results for VVER-440 are now being analyzed. In summary, the consequences of the transient are far milder in the case of VVER-1000.

4. Conclusions

A comparative study of a "restart of the first reactor pump" scenario in both VVER reactor types is now being performed at RCR. This paper presents the first results of DYN3D transient simulations of this boron dilution accident. Future work will focus on further clarification of the scenario and assumptions involved for both reactors, as well as on refinement of the numerical simulation and the calculation procedure. The next step will be to perform simulations of the case with the most reactive control rod stuck out. The burnup version of DYN3D will be used to perform core follow calculations for VVER-440 (Unit 2) and VVER-1000 at Kozloduy NPP (Unit 5) in order to simulate different initial states. Detailed comparisons and analysis of obtained results will complete the first part of this study.

In order to obtain "best estimate" results for this accident scenario, thermal-hydraulic system codes coupled with 3-D reactor core models have to be used. DYN3D has been coupled with the advanced thermal-hydraulic code ATHLET[12]. More realistic transient simulations for both reactors and selected cases will be performed using the coupled code. In conclusion, this study involves multi-level comparisons of the core response of VVER-440 and VVER-1000 for different versions of the original scenario and with different modeling features which will provide a basis for better understanding of the mechanisms behind such accident.

References

1. P. Siltanen and M. Antila, "Extended Protection Against Reactivity Accidents Caused by Slugs Diluted Water in the Loviisa Reactors," Proc. of the second Simposium of AER, p. 327, Paks (1992).

2. P. Siltanen, M. Antila, and R. Kyrki-Rajamaki, "Results of Core Response in Reactivity Accidents due to Local Boron Dilution," Proc. of the fourth Simposium of AER, p. 683, Sozopol (1994).

3. U. Grundmann and U. Rohde, "Influence of a Mixing Model on a Boron Dilution Transient in the VVER-440 Reactor by help of the Code DYN3D/M2," Proc. of the third Simposium of AER, p. 491, Piestany (1993).

4. U. Grundmann and U. Rohde, "Investigations of a Boron Dilution Accident for a VVER-440 Type Reactor by the Help of the Code DYN3D," *Proc. of ANS 1994 Topl. Meeting on Advances in Reactor Physics, Vol. 3, Knoxville (1994).*

5. U. Grundmann and U. Rohde, "DYN3D/M2 - a Code for Calculation of Reactivity Transients in Cores with Hexagonal Geometry," FZR 93 - 01, Rossendorf (1993).

6. U. Rohde, I. Elkin, and V. Kalinenko, "Analysis of a Boron Dilution Accident for VVER-440 Combinig the Use of the codes DYN3D and SITAP," Sahrestagung Kerntechnik'95, p. 111 (1995).

7. S. Mittag, "Solution of AER Benchmark Problem on Control Rod Worth of Paks VVER-440 by the code DYN3D," Proc. of the fourth Symposium of AER, 443, Sozopol (1994).

8. G.Agthe and H. -J. Kretzschmar, "Automated Generation of Neutronics Few-Group Parameters to be used in Macroscopic Computations of Thermal Reactor Cores," (in German), Kernenergie 33, p. 165 (1990).

9. I. Tinka, "VVER Diffusion Data Libraries Prepared by the KASSETA Code", Proc. of the third Symposium of AER, p. 42, Pieštany (1993).

10. "In-Core Fuel Management Code Package Validation for VVERs," IAEA-TECDOC (in print), Vienna.

11. V. Kalinenko, "Analysis of Transients for NPP with VVER-440 Using the Code SITAP," FZR-54, Rossendorf (1994).

12. U. Grundmann, D. Lucas, and U. Rohde, "Coupling of the Advanced Thermal-Hydraulic Code ATHLET with the Neutron Kinetic Core Model DYN3D," *Proc. of the International Topl. Meeting on VVER Safety*, p. 197, Prague (1995).

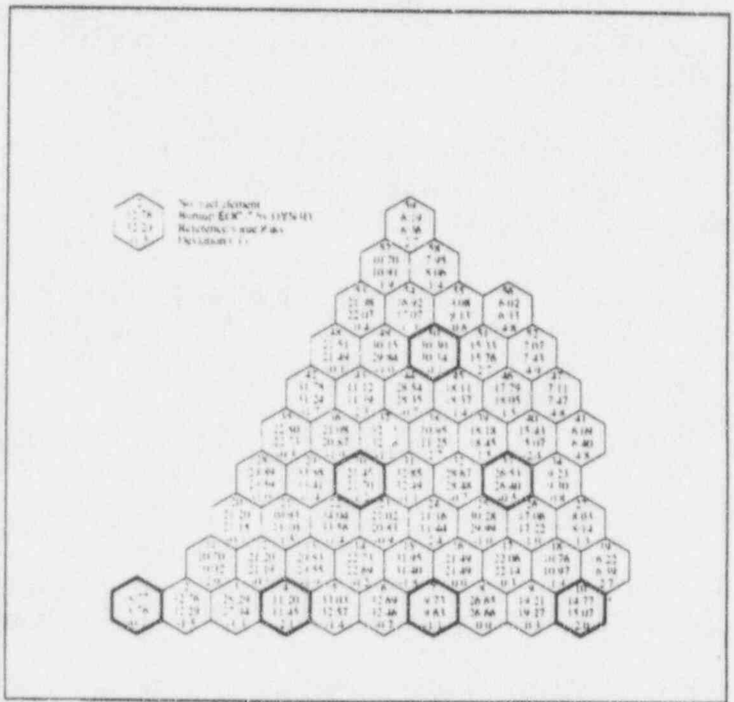


Fig. A. Burnup distribution in MWd/kgU of Paks-2, end of cycle 7; 60-degree sector of core

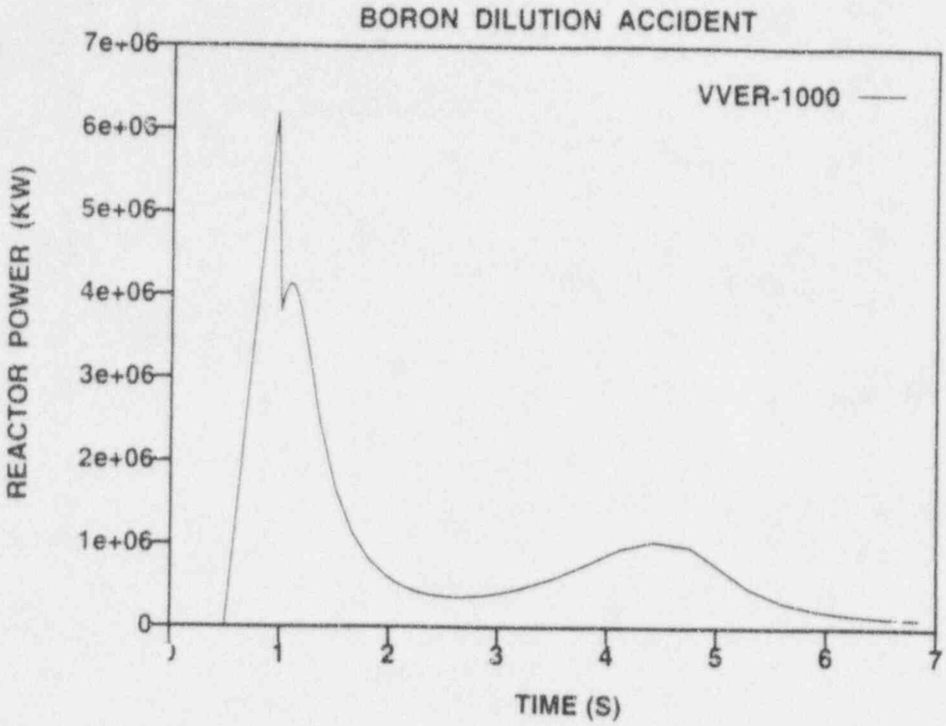


Fig. 1. Reactor Power as a Function of Time.

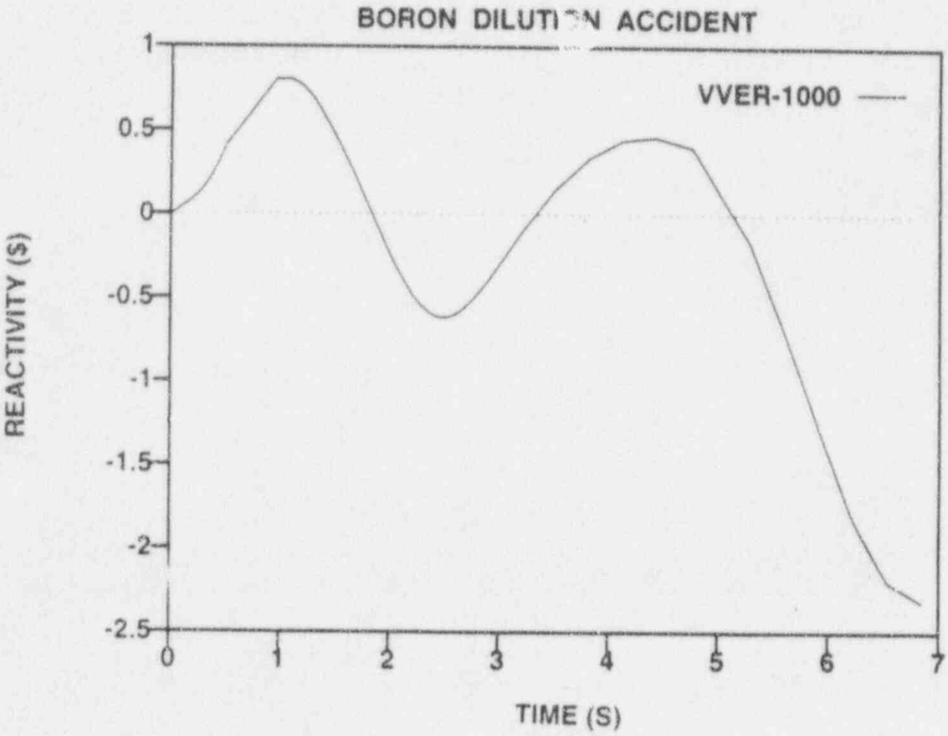


Fig. 2. Reactivity as a Function of Time.

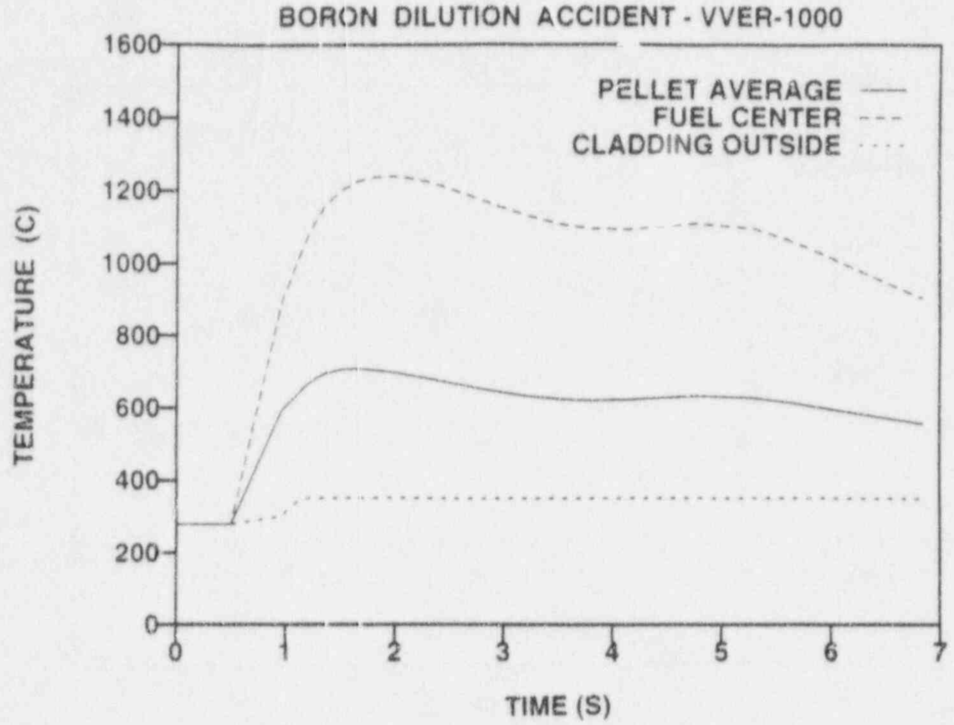


Fig. 3. Maximal Temperatures as a Function of Time.

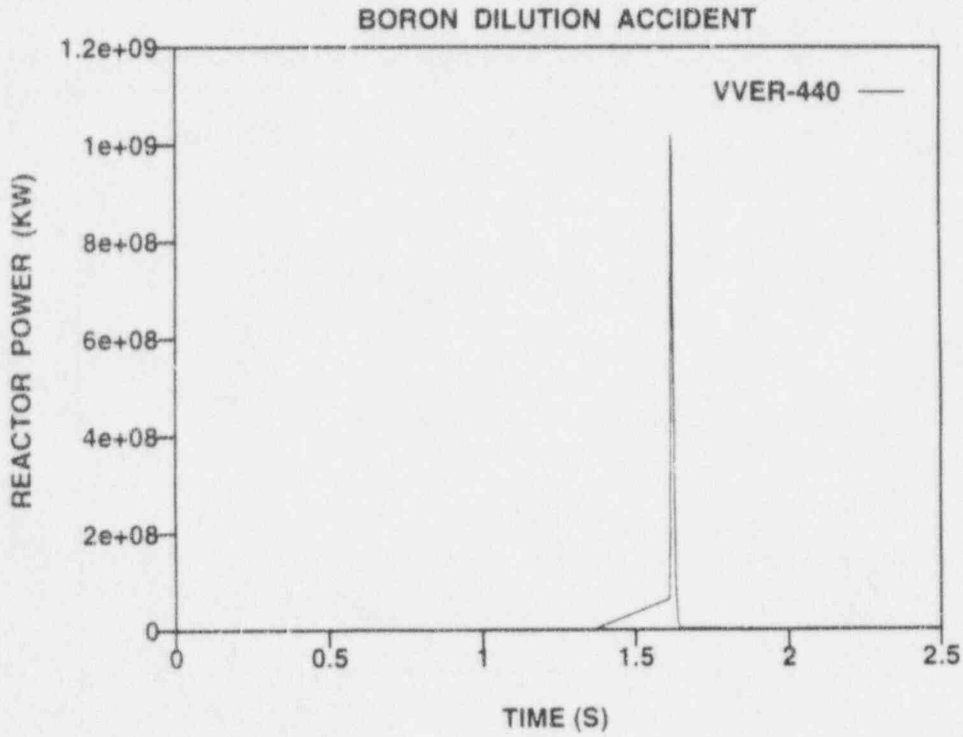


Fig. 4. Reactor Power as a Function of Time.

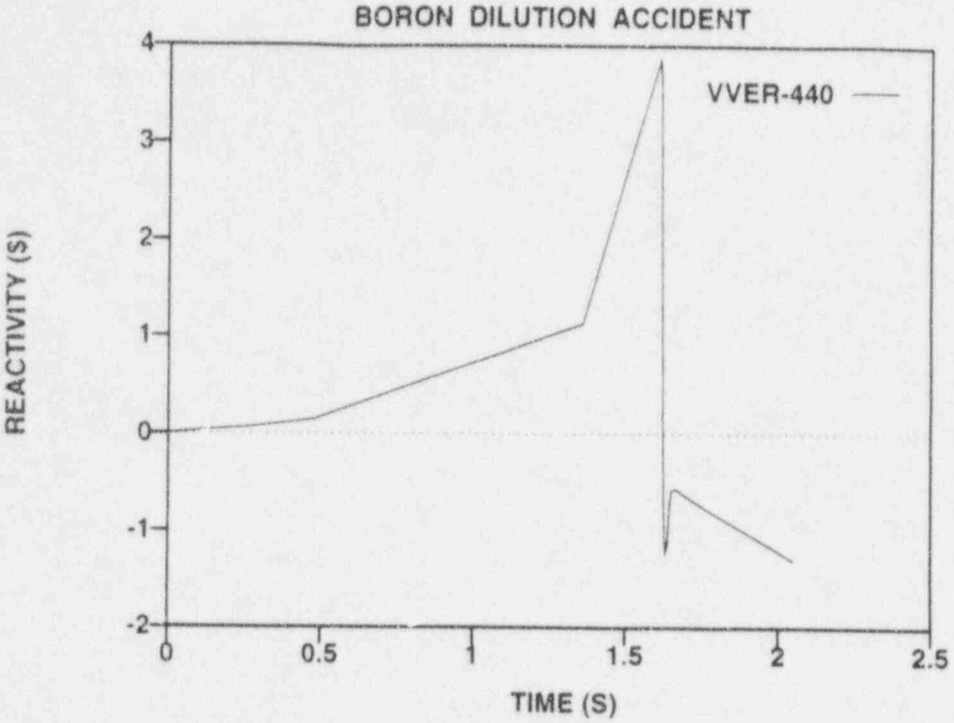


Fig. 5. Reactivity as a Function of Time.

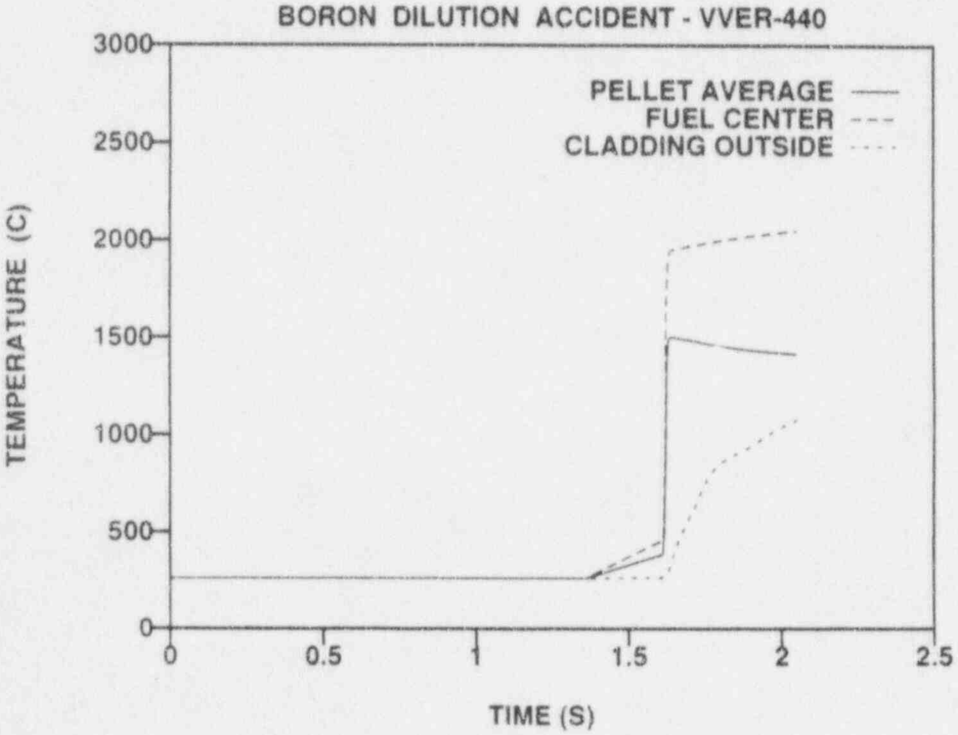


Fig. 6. Maximal Temperatures as a Function of Time.

THE SYSTEM 80+™ RESPONSE TO THE SMALL BREAK LOCA BORON DILUTION ISSUE

J. Longo, Jr., F. L. Carpentino, R. C. Mitchell

ABB Combustion Engineering Nuclear Systems
A Business Unit of Combustion Engineering, Inc.
Windsor, Connecticut, USA

ABSTRACT

The small break loss of coolant accident (LOCA) boron dilution issue surfaced in late 1992 as a result of concerns raised by analyses performed by the Finnish Centre for Radiation and Nuclear Safety (STUK)[1]. The concern is the potential effect on criticality and fuel temperature if the unborated condensate that is formed during the event is transported to the core. ABB-Combustion Engineering (ABB-CE) addressed this new issue during the U.S. Nuclear Regulatory Commission (USNRC) review of the System 80+™ Advanced Light Water Reactor (ALWR) design[2]. The USNRC approved the ABB-CE response to this postulated event and issued a Final Design Approval (FDA) for the System 80+ design in July 1994[3].

The small break LOCA (SBLOCA) boron dilution issue addresses the potential large accumulation of unborated condensate in the reactor coolant system (RCS) piping following a SBLOCA. This condensate is postulated to be suddenly introduced into the core by restarting a reactor coolant pump (RCP) or by regaining natural circulation. The unborated condensate is assumed to accumulate as the result of boiling in the core and condensing in the steam generator (SG) tubes. This condensate may collect in the RCP suction piping and then later may be pushed into the reactor vessel where it could lead to an increase in reactivity.

The small break LOCA boron dilution event has been studied extensively for the System 80+ design. This paper summarizes how a combination of design features, improved operating guidance, and analysis have been used to resolve this issue. The design features minimize the amount of condensate which can be formed and minimize the boron concentration required to prevent criticality. The operating guidance significantly reduces the probability of an incorrect RCP restart. Finally, analyses show that adequate core cooling is maintained even if conservative assumptions are applied.

INTRODUCTION

There is a span of time during a small break LOCA when steam is generated in the core and is condensed in the steam generator tubes. The steam can be postulated to be largely devoid of the boric acid, which remains dissolved in the highly borated liquid in the core. Thus, the resulting condensate that collects in the RCP suction piping may be largely devoid of dissolved boric acid.

A portion of the condensate runs down the upflow side of the steam generator tubes and returns to the reactor vessel (Figure 1). This process, called reflux boiling, transfers heat from the core to the steam generators when the primary side liquid level falls below the tops of the steam generator tubes so that natural circulation of the RCS liquid ceases. Condensation in the steam generators continues so long as the primary side temperature exceeds that of the secondary side. The remaining portion of the condensate runs down the downflow side of the steam generator tubes and collects in the cold leg side of the RCS, the lowest portions of which are the loop seals in the suction leg piping between the steam generator outlet plena and the RCPs.

If the unborated condensate, which has accumulated in the RCP suction piping, is assumed to return to the reactor vessel without sufficient mixing with the highly borated inventory in the reactor downcomer and lower plenum, the resulting reduced boron concentration of the slug of coolant entering the core may result in a return to a critical condition. A critical core will result in neutron power generation which will affect the fuel rod (clad) temperature. The condensate in the RCP suction piping may be driven into the core by restarting a RCP or by the re-establishment of natural circulation in the RCS.

A detailed survey of SBLOCA analyses has been conducted to determine which cases could produce a significant amount of condensate. The survey has identified that, as a result of the System 80+^{TM} ALWR four train safety injection system (SIS) design, a very narrow range of break sizes have this potential (equivalent break diameters of one to three inches). Smaller breaks regain inventory too quickly to accumulate much condensate and larger breaks remove the steam to the containment with little or no condensation in the SGs. The reactivity response of the core has been addressed for this narrow range of breaks by assuming a conservative amount of condensate for these cases and analyzing two basic scenarios.

One scenario assumes that the condensate exceeds the core volume, remains completely unborated, and is introduced to the core at a rate associated with regaining RCS natural circulation. This analysis demonstrates that the shutdown margin of the System 80+ design is sufficient to minimize the return to power and preserve adequate core cooling. The resulting return to power is derived conservatively and it is still shown that the peak fuel cladding temperatures remain well below acceptable limits. Separate mixing calculations show that there is significant mixing between the unborated water and the highly borated water in the vessel such that the core would remain subcritical.

A second scenario assumes that the condensate volume associated with the piping of two RCPs is rapidly injected to the reactor vessel. Although conditions for RCP restart were already stated in the System 80+ Emergency Operating Guidelines (EOGs), the EOGs were revised by plant operations and human factors experts to substantially reduce the probability of incorrect restart. In addition, a detailed multi-dimensional analysis of mixing has been performed, assuming an undesirable restart of a RCP, to determine the lowest local boron concentration that could occur in the core. The mixing calculation has also been done conservatively by varying the amount of unborated condensate and by ignoring several potentially significant contributions to mixing. This bounding assessment shows that the lowest local boron concentration in the core (1850 ppm) significantly exceeds the minimum required concentration (550 ppm) to maintain the core subcritical even at cold conditions (300 F, 150 C). Therefore, although incorrect RCP restart is very unlikely, the core will remain subcritical and adequately cooled even if it occurred.

DESIGN FEATURES WHICH MINIMIZE EVENT SEVERITY

For small break sizes, the RCS refills very quickly and very little condensate is formed. For large break sizes, the RCS may not refill and condensation is minimized by removing most or all of the core heat through the break. This will minimize the condensate available to be delivered to the core should an RCP be restarted. The break sizes of concern range from one to three inches in diameter (2.5 cm to 7.6 cm). For this range of break sizes, assuming operator cooldown of the SG secondary side starting at one half hour after the inception of the SBLOCA and at the maximum cooldown rate permitted by the EOGs, the condensate volume available per cold leg is small. This condensate would contain some boron because of steam transport, liquid entrainment, SIS backflow, and some of the initial inventory. The boric acid concentration inside the core support barrel at RCS refill time is very high as a result of the extended period of boiling of the highly borated (4000 ppm) SIS water. The boron concentration in the core support barrel ranges from 21,500 to 25,500 ppm.

The effect of the critical boric acid concentration and transient xenon effects on a hypothetical return to power after a SBLOCA has been investigated. The System 80+™ design has a control rod pattern which provides complete coverage of all assemblies. For post-LOCA conditions, boric acid concentrations greater than 550 ppm will maintain the core subcritical at beginning of cycle and 300 F (150 C). Zero ppm will maintain the core subcritical after the first third of the fuel cycle. If xenon effects are accounted for, less than 20 percent of the cycle life would have any risk of post-LOCA return to power while near peak xenon.

The Nuplex 80+™ Advanced Control Complex also provides design features which help the operators to rapidly and accurately diagnose the LOCA event. The Nuplex 80+ design takes advantage of modern digital processing equipment to implement the protection, control, and monitoring systems. Nuplex 80+ uses critical function monitoring algorithms and setpoints designed specifically for the LOCA event to best support the operators' recovery actions as defined in the Emergency Operating Guidelines. Success path monitoring provides real time status of each critical function success path. The result is improved situational awareness for the operating staff which reduces the risk of improper action.

EOG MODIFICATIONS TO MINIMIZE RISK OF INCORRECT RCP RESTART

Based on guidance from plant operators and human factors experts, the System 80+™ Emergency Operating Guidelines (EOGs) were modified to reduce the likelihood of an erroneous restart of an RCP following a LOCA. To ensure that the operator does not believe that starting a RCP is more important than verifying natural circulation, the existing guidelines were re-ordered as follows:

1. Verify adequate single-phase natural circulation.
2. If single-phase natural circulation cannot be established, verify adequate two phase natural circulation.
3. Determine if RCP restart is needed and desired.
4. Verify that all RCP restart criteria are met.
5. Restart RCP.

Supplemental information was added to the LOCA guideline that cautions the operator about the possibility of condensate buildup in the RCP suction piping prior to making the decision to start a RCP. Step 3 was revised to include direction from the Technical Support Center (TSC) by evaluation of the RCP restart desirability. Step 4 was revised to require the operator to obtain permission from the TSC prior to starting a RCP.

The operator will reach these steps after attempting to isolate the leak. First, the step to verify adequate single-phase natural circulation will be reached. If adequate single-phase natural circulation exists, the operator will skip the next step and proceed to the step to determine the desirability of restarting an RCP. If the operator cannot verify adequate single-phase natural circulation, the operator will proceed to the next step which verifies adequate two-phase natural circulation.

Once the "RCP restart" steps are reached, the caution will be read alerting the operator to the possibility of condensate buildup in the RCP suction piping. Next, the operator will request that the TSC determine the desirability and need to restart an RCP. The operator and TSC will base their decision on at least the following criteria: adequacy of core heat removal using natural circulation, the need for main pressurizer sprays, existing RCS pressure and temperature, duration of component cooling water interruption to the RCPs, RCP seal conditions, time the plant was in two-phase natural circulation, time the plant was in single-phase natural circulation. If any of these criteria do not indicate that an RCP restart is desirable, the operator will skip the remaining steps. Otherwise, the operator will proceed to the next step.

The operator next determines if an RCP restart can be performed based on at least the following criteria: the TSC has recommended RCP restart, single-phase natural circulation has been established for at least 20 minutes, power is available to the RCP bus, RCP auxiliaries are operating, at least one SG is available, pressurizer level is acceptable, RCS is subcooled, RCP operating instruction criteria are satisfied. Twenty minutes of natural circulation allows more than two complete RCS loop circuits. Plant tests have shown that significant mixing will be accomplished in that time.

If any of the above criteria do not indicate that an RCP restart is desirable, the operator will skip the remaining steps. Otherwise, the operator will proceed to the next step and start a RCP.

A Human Reliability Analysis (HRA) has been performed to determine the probability of erroneously restarting an RCP prior to the establishment of at least 20 minutes of continuous single-phase natural circulation. The following conservative assumptions have been made: the control room staff will always want to restart a RCP (contrary to operating experience), no plant conditions exist which preclude a RCP restart other than the lack of 20 minutes of single-phase natural circulation, no credit is taken for improved plant or procedure ergonomics, and the activities required for pump restart require zero time. Three other reasonable assumptions have been made: no single random error will cause a RCP restart, the TSC will proactively communicate if asked about pump restart, and the Main Control Room will follow the TSC directions. Based on these assumptions, the Human Error Probability of erroneously restarting a RCP given a small break LOCA occurrence has been determined to be 3.9 E-4.

ANALYSES OF THE SBLOCA BORON DILUTION EVENT

Analyses of Boron Dilution Resulting From Natural Circulation Restart

As natural circulation is re-established, condensate will flow from the RCP suction piping toward the reactor vessel. Mixing between the relatively unborated condensate and borated safety injection system (SIS) liquid occurs in the downcomer and reactor vessel lower head.

A bounding analysis has been performed for a conservatively selected cold side break of 0.005 square feet. This small break size is selected because the time of RCS refill is early and therefore the decay heat which drives the natural circulation is higher. No credit is taken for any of the mixing of borated and unborated water that is expected to occur in the downcomer and lower plenum of the reactor vessel. Instead, the condensate is assumed to enter the core as a slug of pure water moving at a natural circulation flow rate consistent with that of a small cold leg side break at the time of RCS refill. The size of the unborated slug is assumed to be unlimited. That is, unborated condensate is assumed to continue to enter the bottom of the core for the duration of the calculated transient.

The effective size of the slug that has entered the core up to the end of the calculated transient is greater than that of the core liquid volume.

The analysis consists of iteratively performing thermal hydraulic analyses and core physics analyses. The RCS thermal hydraulic analyses are performed to determine the change in pressure and natural circulation flow rate as the unborated slug enters the core and core-to-coolant heat transfer takes place. The core physics analyses are performed to determine reactivity, power peaking, and transient power. These analyses show that the core would return to a critical condition when the unborated slug has progressed into the core. As the slug progresses further into the core, the resultant neutron power would experience a very brief spike which would be terminated by Doppler feedback in the fuel. The power would then drop further as coolant heatup results in moderator density reactivity feedback. The power would undergo several oscillations of diminishing amplitude and finally settle at a level that is a small fraction of full power. The neutron power is shown in Figure 2. If the analysis had accounted for borated water entering the core behind the slug, then the power would have rapidly decreased towards zero.

The transient analysis of the hot rod shows that the peak clad temperature would remain fairly low because the fuel rod surface remains in a pre-DNB heat transfer mode. In addition, the fuel centerline temperature at the hot spot would remain well below the melting value. These temperatures are shown in Figure 3.

The analyses reveal that boron dilution during a small break LOCA, even when calculated using the highly conservative assumption of an unborated slug of water entering the core, results in fuel rod conditions which are well below the acceptance criteria for the Emergency Core Cooling System (ECCS).

Analyses of Boron Dilution Resulting From RCP Restart

Restart of a RCP could introduce unborated water into the core at an unacceptably high rate if no mixing would occur. Analyses have been performed to show that, even in the highly unlikely case of a RCP restart at the worst time, sufficient mixing occurs to ensure that the core remains subcritical and adequately cooled.

To envelope possible back flow from the adjacent RCP suction pipe which is connected through the common SG plenum, the volume of the slug of unborated water assumed to be injected into the reactor vessel by the start of one RCP is taken to be twice the volume accumulated in one RCP suction pipe. The volume assumed is conservative because it exceeds the amount of condensate than can be generated in the time allotted.

A computational fluid dynamic analysis has been performed using the FLUENT code[4]. Even with the conservative estimate of the volume of the unborated water, the FLUENT analysis demonstrates substantial mixing in the lower annulus and lower head of the reactor vessel with the start-up of one RCP. A two-dimensional axisymmetric model (radial plane) of the reactor vessel from the top of the fuel alignment plate to the bottom of the lower head is applied to model the turbulent mixing of chemical species. The grid structure allows for the calculation of the downcomer pressure drop and radial mixing. It also allows for the calculation of the lower head turning losses and associated shear-generated turbulence and mixing. The inlet velocity boundary condition at the downcomer is time dependent to reflect the pump flow rate acceleration. The calculation shows that the mixing is sufficient to ensure that the core remains adequately borated to remain subcritical. Figure 4 shows the transient boron concentration.

CONCLUSION

The issue of SBLOCA boron dilution has been studied for the System 80+™ plant design. The System 80+ plant design minimizes the amount of condensate that can be formed. Any condensate will be mixed with highly borated water in the RCS upon the onset of single-phase natural circulation or restart of a RCP. The core design requires less than 550 ppm boron at 300 F (150 C) to remain subcritical over the first third of core life. No boron is required over the remaining two thirds of core life.

Modifications have been made to the EOGs to significantly reduce the probability of an incorrect restarting of a RCP prior to achieving adequate mixing by natural circulation. The probability of an operator erroneously restarting a RCP is also reduced because of the advanced diagnostics and monitoring in the Nuplex 80+™ advanced control room.

For boron dilution resulting from the restart of natural circulation, a core coolability assessment has been performed assuming no mixing of the condensate with the highly borated water in the reactor vessel. The results of this very conservative, bounding analysis show that even if an arbitrarily large slug of totally unborated coolant is assumed to pass through the core, the peak clad temperature and oxidation values remain well below the limits of the ECCS acceptance criteria. Separate mixing calculations show that there is significant mixing between the unborated water and the highly borated water in the vessel such that the core would remain subcritical.

Even if the restart of an RCP is assumed to occur at the worst time, sufficient mixing occurs to ensure that the core remains subcritical and adequately cooled. The minimum boron concentration in the core is calculated to be greater than 1850 ppm. This is well above the value required to prevent criticality.

Thus, the SBLOCA boron dilution issue is not a problem for the System 80+ design. The plant design features, EOGs, and the boron mixing demonstrated by analyses combine to eliminate concerns for this event.

REFERENCES

1. J. Hyvarinen, "An Inherent Boron Dilution Mechanism in Pressurized Water Reactors," Finnish Centre for Radiation and Nuclear Safety (STUK)

2. The Combustion Engineering Standard Safety Analysis Report Design Certification (CESSAR DC), Docket No. 52-002

3. Final Safety Evaluation Report Related to the Certification of the System 80+ Design, NUREG-1462, August, 1994

4. FLUENT User's Guide Version 4.0, Fluent, Inc., December, 1991

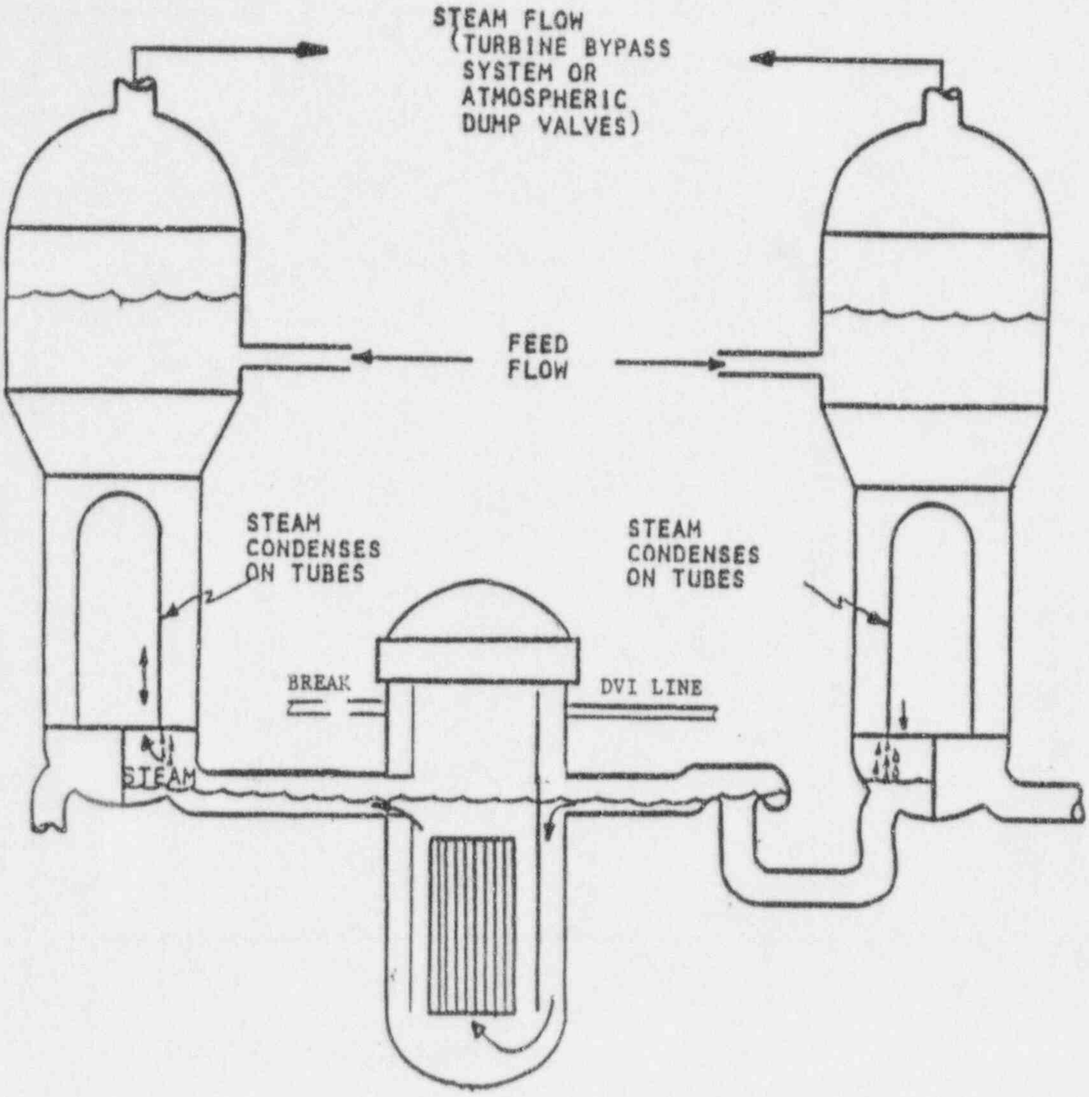


Figure 1
Heat Removal Via Small Break
(Reflux Cooling)

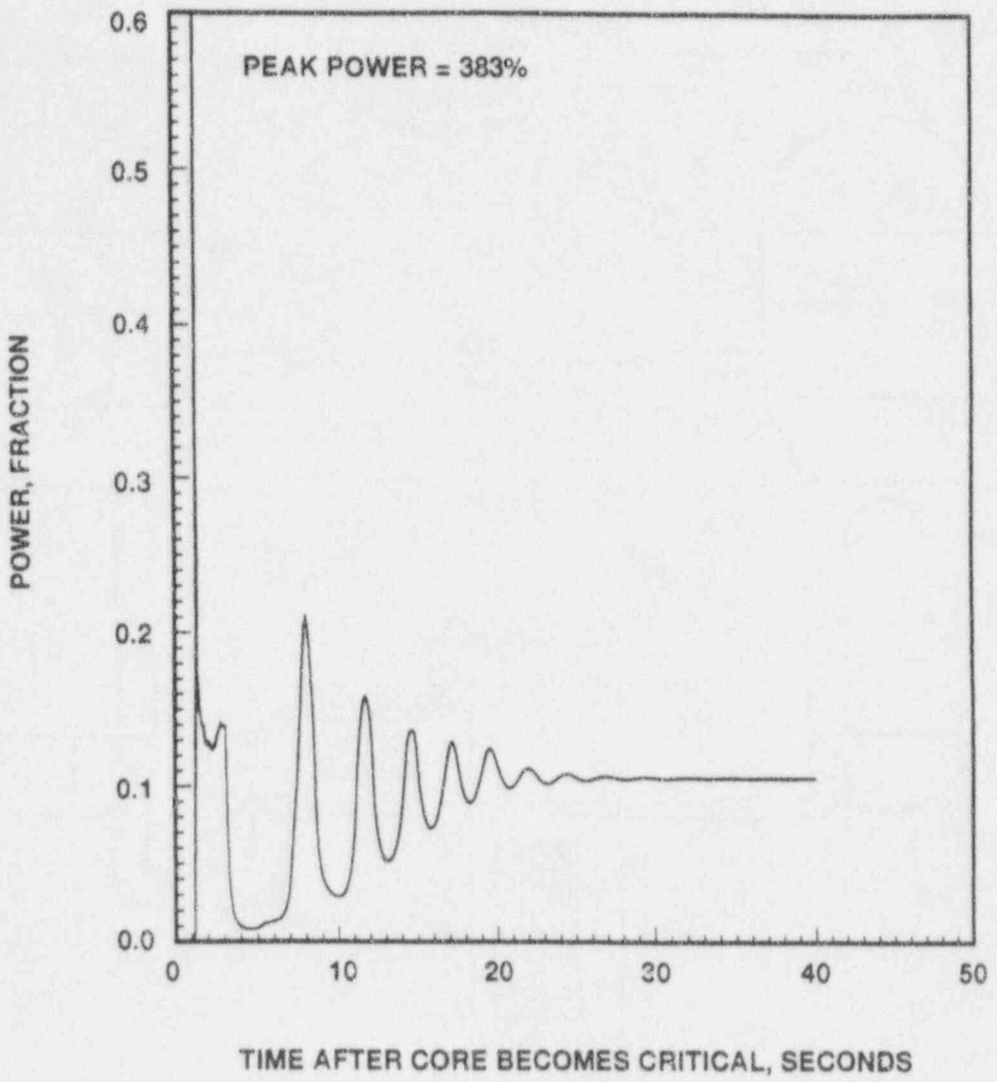


Figure 2
Normalized Power After Core Becomes Critical

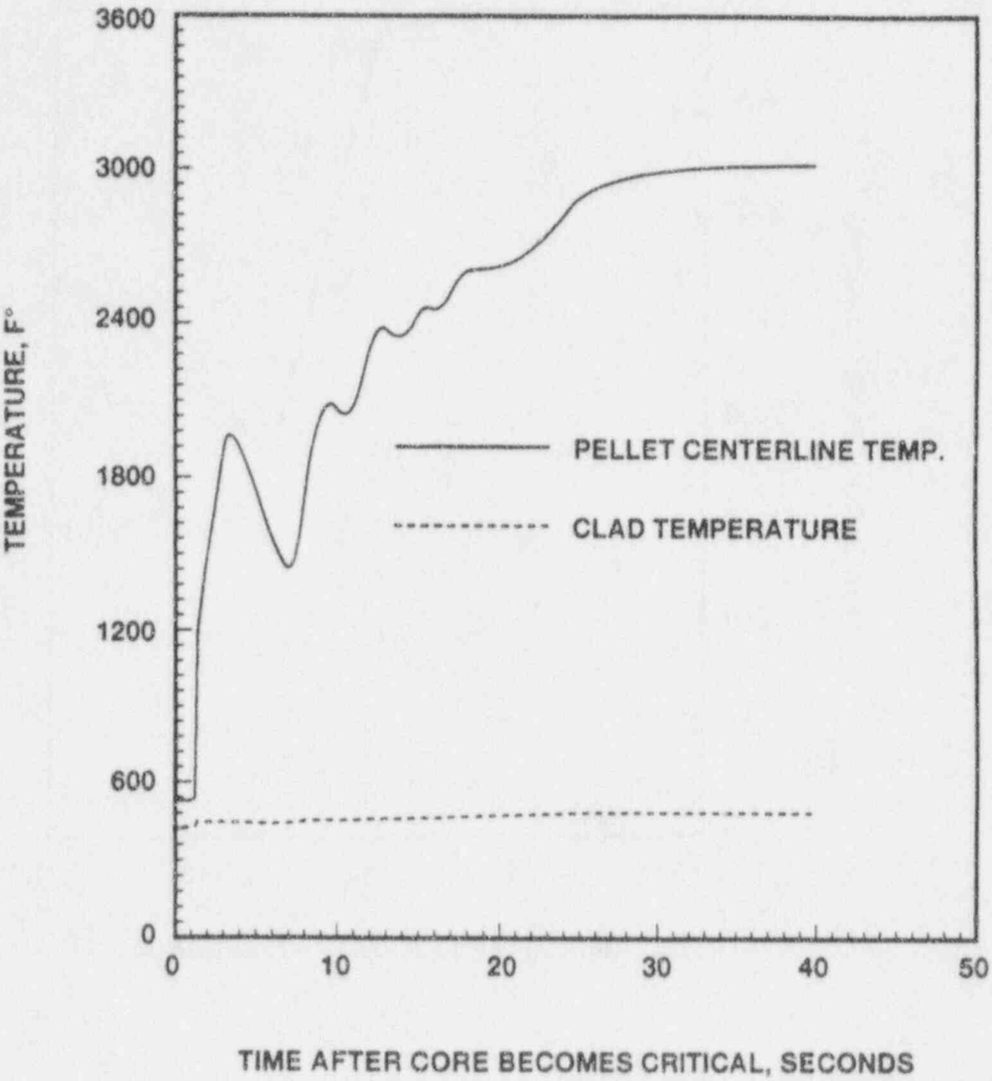


Figure 3
Hot Spot Temperature Transients

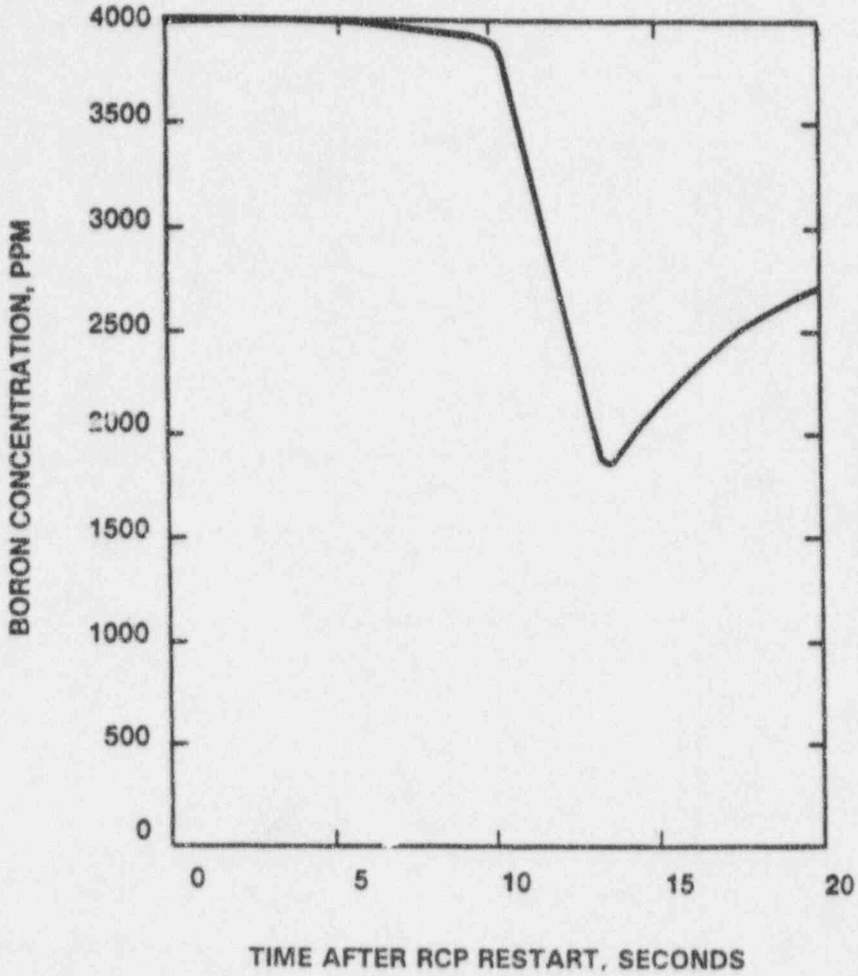


Figure 4
Transient Boron Concentration with 1-Pump Restart

EXPLANATION OF CORE PHYSICS PHENOMENA DURING A BORON DILUTION TRANSIENT

C. J. Jackson and H. Finnemann

Siemens AG Power Generation Group (KWU)
P.O. Box 3220
D–91050 Erlangen
Germany
Tel. 49–9131–18–2317

ABSTRACT

As an aid to help explain core physics phenomena in a neutron kinetics calculation, a reactivity edit option for the coupled RELAP5/PANBOX 2 code system has been developed. This option calculates total core reactivity at each time step, as well as reactivity contributions from sources such as boron concentration, fuel temperature, moderator density, and neutron flux shape. The results of a calculation of a hypothetical boron transient are also presented. These results indicate that changes in the neutron flux shape could be a very strong stabilizing factor in some transients. Thus, it is important to calculate such transients with a multidimensional neutron kinetics code.

1. INTRODUCTION

Under certain flow conditions, boron dilution transients can exhibit a strong coupling between the plant thermalhydraulics and core neutron kinetics phenomena. The reason for this is twofold: firstly, the core power is the driving force of the coolant flow; and secondly, the local coolant boron concentrations have a direct influence on the core reactivity. With this strongly coupled system, it is very useful to be able to calculate plant thermalhydraulic and core physics phenomena at the same time. To accomplish this, we have coupled the best estimate light water system analysis code RELAP5[1] with the core simulator PANBOX 2[2]. The resulting code system[3,4] enables us to calculate plant transients, such as a boron dilution transient, with a three–dimensional neutron kinetics model.

The standalone RELAP5 code utilizes the simple point kinetics approximation to model the core behavior during a transient. Coupling between the core and the thermalhydraulic system is normally achieved by the input of fuel temperature, moderator density and boron concentration coefficients. In contrast, with the RELAP5/PANBOX 2 code, coupling between the core and the thermalhydraulic system is accomplished via the dependence of the macroscopic cross sections on local thermalhydraulic conditions and boron concentrations. The point kinetics model produces results which are intuitively understandable: the contributions to the reactivity from fuel temperature, coolant density, and boron density are readily obtained and interpreted. In contrast, when the reactor kinetics are calculated using a three–dimensional multigroup neutron diffusion model, these reactivity contributions are no longer apparent . Thus, we found that the incorporation of a three–dimensional kinetics model into RELAP5 created a new challenge for the analyst to explain transient phenomena, despite the fact that the transient could be calculated with greater accuracy.

To help explain transient phenomena, we have developed a 'reactivitiy edit option' for PANBOX 2. The option uses the neutron flux distribution at each time point to calculate the total core reactivity, as well as the contributions to this reactivity from changes in the core state. These changes include changes in fuel temperature, changes in moderator density, and changes in neutron flux distribution. In section 2, we define and derive these contributions from a strict theoretical point of view. In section 3, we describe how we calculate the contributions in our nodal kinetics code PANBOX 2. Finally we illustrate our method in section 4, and draw some conclusions in section 5.

2. DEFINITION OF REACTIVITY AND REACTIVITY CONTRIBUTIONS

We begin with the general definition of reactivity.[5]

$$\varrho(t) = \frac{< W(\vec{r}, E), \mathbf{T}(\vec{r}, E, t) S(\vec{r}, E, t) >}{< W(\vec{r}, E), \mathbf{P}(\vec{r}, E, t) S(\vec{r}, E, t) >} \tag{1}$$

where **P** is the fission source operator, and **T** is the combined diffusion–theory operator,

$$\mathbf{T}(\vec{r}, E, t) \equiv \mathbf{P}(\vec{r}, E, t) - \mathbf{A}(\vec{r}, E, t) - \mathbf{L}(\vec{r}, E, t) \tag{2}$$

The operator **A** contains absorption and scattering terms and the operator **L** accounts for leakage effects. W is some generalized weighting function, S is the neutron flux shape, and <•> denotes integration over all space and energy. The neutron flux shape is normalized from the neutron flux through

$$S(\vec{r}, E, t) = \frac{\phi(\vec{r}, E, t)}{\langle W(\vec{r}, E), \phi(\vec{r}, E, t) \rangle} \tag{3}$$

For an initial flux distribution ϕ_0, which satisfies the eigenvalue problem

$$\left[(1 - \varrho_0)\mathbf{P}_0(\vec{r}, E) - \mathbf{A}_0(\vec{r}, E) - \mathbf{L}_0(\vec{r}, E)\right]\phi_0(\vec{r}, E) = 0 \tag{4}$$

(with specified boundary conditions), equation (1) takes the form

$$\varrho_0 = \frac{< W(\vec{r}, E), \mathbf{T}_0(\vec{r}, E) S_0(\vec{r}, E) >}{< W(\vec{r}, E), \mathbf{P}_0(\vec{r}, E) S_0(\vec{r}, E) >} \tag{5}$$

which is stationary for all choices of W which are non–zero. During a transient calculation, the operators **P**, **A**, and **L** all change due to changes in neutron cross sections. These changes can be expressed as contributions from various sources. For example,

$$\Delta\mathbf{P}(\vec{r}, E, t) \equiv \mathbf{P}(\vec{r}, E, t) - \mathbf{P}_0(\vec{r}, E) = \Delta\mathbf{P}_{ppm} + \Delta\mathbf{P}_{ft} + \Delta\mathbf{P}_{mt} + \Delta\mathbf{P}_{md} + \Delta\mathbf{P}_{other} \tag{6a}$$

$$\Delta\mathbf{A}(\vec{r}, E, t) \equiv \mathbf{A}(\vec{r}, E, t) - \mathbf{A}_0(\vec{r}, E) = \Delta\mathbf{A}_{ppm} + \Delta\mathbf{A}_{ft} + \Delta\mathbf{A}_{mt} + \Delta\mathbf{A}_{md} + \Delta\mathbf{A}_{other} \tag{6b}$$

$$\Delta\mathbf{L}(\vec{r}, E, t) \equiv \mathbf{L}(\vec{r}, E, t) - \mathbf{L}_0(\vec{r}, E) = \Delta\mathbf{L}_{ppm} + \Delta\mathbf{L}_{ft} + \Delta\mathbf{L}_{mt} + \Delta\mathbf{L}_{md} + \Delta\mathbf{L}_{other} \tag{6c}$$

where the subscripts stand for changes in boron ppm, changes in fuel temperature, changes in moderator temperature, changes in moderator density, and other changes, respectively. Our goal is to identify how operator changes contribute to the change in reactivity.

The total change in reactivity since the initial condition is simply equation (5) subtracted from equation (1):

$$\Delta\varrho(t) \equiv \varrho(t) - \varrho_0 = \frac{< W(\vec{r}, E), \mathbf{T}(\vec{r}, E, t) S(\vec{r}, E, t) >}{< W(\vec{r}, E), \mathbf{P}(\vec{r}, E, t) S(\vec{r}, E, t) >} - \frac{< W(\vec{r}, E), \mathbf{T}_0(\vec{r}, E) S_0(\vec{r}, E) >}{< W(\vec{r}, E), \mathbf{P}_0(\vec{r}, E) S_0(\vec{r}, E) >} \tag{7}$$

The expression on the right hand side is conveniently divided into four main contributions defined as

$$\Delta\varrho(t) = \Delta\varrho_{\Delta\phi}(t) + \Delta\varrho_{\Delta T}(t) + \Delta\varrho_{\Delta T\Delta\phi} + \Delta\varrho_{\gamma}(t) \tag{8}$$

$$\Delta\varrho_{\Delta\phi}(t) \equiv \frac{< W(\vec{r}, E), \mathbf{T_0}(\vec{r}, E)S(\vec{r}, E, t) >}{< W(\vec{r}, E), \mathbf{P_0}(\vec{r}, E)S(\vec{r}, E, t) >} - \frac{< W(\vec{r}, E), \mathbf{T_0}(\vec{r}, E)S_0(\vec{r}, E) >}{< W(\vec{r}, E), \mathbf{P_0}(\vec{r}, E)S_0(\vec{r}, E) >} \tag{9a}$$

$$\Delta\varrho_{\Delta T}(t) \equiv \frac{< W(\vec{r}, E), \Delta\mathbf{T}(\vec{r}, E, t)S_0(\vec{r}, E) >}{< W(\vec{r}, E), \mathbf{P}(\vec{r}, E, t)S(\vec{r}, E, t) >} \tag{9b}$$

$$\Delta\varrho_{\Delta T\Delta\phi}(t) \equiv \frac{< W(\vec{r}, E), \Delta\mathbf{T}(\vec{r}, E, t)\Delta S(\vec{r}, E, t) >}{< W(\vec{r}, E), \mathbf{P}(\vec{r}, E, t)S(\vec{r}, E, t) >} \tag{9c}$$

$$\Delta\varrho_{\gamma}(t) \equiv \gamma\frac{< W(\vec{r}, E), \mathbf{T_0}(\vec{r}, E)S(\vec{r}, E, t) >}{< W(\vec{r}, E), \mathbf{P_0}(\vec{r}, E)S(\vec{r}, E, t) >} \tag{9d}$$

where

$$1 + \gamma \equiv \frac{1}{1 + \frac{< W(\vec{r},E),\Delta\mathbf{P}(\vec{r},E,t)S(\vec{r},E,t) >}{< W(\vec{r},E),\mathbf{P_0}(\vec{r},E,t)S(\vec{r},E) >}} \tag{10}$$

The meaning of the first three terms is clear: (9a) is the reactivity change due only to changes in the neutron flux shape, (9b) is the reactivity change due to direct changes in cross sections, and (9c) are the combined changes due to changes in cross section and flux shape. The meaning of the $\Delta\varrho_\gamma$ term will become clear once we have chosen a suitable weight function W.

The goal in choosing the weight function is to make the different reactivity components (9a) to (9d), at least as strongly dependent on the changes in operators as on the changes in flux. We see directly from (9b) and (9c) that $\Delta\varrho_{\Delta T}$ and $\Delta\varrho_{\Delta T\Delta\phi}$ already have this strong dependence on ΔT. Expanding (10) in a series, the leading terms of γ are

$$\gamma = -\frac{< W(\vec{r}, E), \Delta\mathbf{P}(\vec{r}, E, t)S(\vec{r}, E, t) >}{< W(\vec{r}, E), \mathbf{P_0}(\vec{r}, E)S(\vec{r}, E) >} + \frac{< W(\vec{r}, E), \Delta\mathbf{P}(\vec{r}, E, t)S(\vec{r}, E, t) >^2}{< W(\vec{r}, E), \mathbf{P_0}(\vec{r}, E)S(\vec{r}, E) >^2} - \dots \tag{11}$$

which shows that $\Delta\varrho_\gamma$ is dependent on $\Delta\mathbf{P}$ to first order. Only (9a) does not have this first order dependence on the change in the operator. The obvious choice for the weight function is therefore the solution to the adjoint of equation (4):

$$\left[(1 - \varrho_0)\mathbf{P_0^*}(\vec{r}, E) - \mathbf{A_0^*}(\vec{r}, E) - \mathbf{L_0^*}(\vec{r}, E)\right]\phi^*(\vec{r}, E) = 0 \tag{12}$$

(Appropriate boundary conditions must also be chosen for the adjoint problem.) When this weight function is used, equation (9a) may be expressed as

$$\begin{aligned}\Delta\varrho_{\Delta\phi}(t) &= \frac{< \mathbf{T_0^*}(\vec{r}, E)\phi^*(\vec{r}, E), S(\vec{r}, E, t) >}{< \mathbf{P_0^*}(\vec{r}, E)\phi^*(\vec{r}, E), S(\vec{r}, E, t) >} - \frac{< \phi^*(\vec{r}, E), \mathbf{T_0}(\vec{r}, E)S_0(\vec{r}, E) >}{< \phi^*(\vec{r}, E), \mathbf{P_0}(\vec{r}, E)S_0(\vec{r}, E) >} \\ &= \varrho_0 - \varrho_0 \\ &= 0\end{aligned} \tag{13}$$

for all nontrivial ϕ^* and S_0. Thus, the use of the adjoint function as a weight function eliminates the first order reactivity contribution due to flux change. Additionally, it casts light on the meaning of $\Delta\varrho_\gamma$, which now may be written as

$$\Delta\varrho_\gamma(t) = \gamma\varrho_0$$
$$= -\varrho_0 \frac{< W(\vec{r},E), \Delta\mathbf{P}(\vec{r},E,t)S(\vec{r},E,t) >}{< W(\vec{r},E), \mathbf{P}_0(\vec{r},E)S(\vec{r},E) >} + O(\varrho_0\Delta\mathbf{P}^2) \quad (14)$$

That is, $\Delta\varrho_\gamma$ may be interpreted as a shift of the initial core reactivity ϱ_0 due to a change in the production operator. When the initial core reactivity is zero, $\Delta\varrho_\gamma$ is also zero.

Now the two main reactivity contributions $\Delta\varrho_{\Delta T}$ and $\Delta\varrho_{\Delta T\Delta\phi}$ may be split up into contributions from the different core physics phenomena. Using (6a) to (6c) in (9b), for example, we define the following different contributions:

$$\Delta\varrho_{ppm}(t) \equiv \frac{< W(\vec{r},E), \Delta\mathbf{T}_{ppm}(\vec{r},E,t)S_0(\vec{r},E) >}{< W(\vec{r},E), \mathbf{P}(\vec{r},E,t)S(\vec{r},E,t) >} \quad (15a)$$

$$\Delta\varrho_{ft}(t) \equiv \frac{< W(\vec{r},E), \Delta\mathbf{T}_{ft}(\vec{r},E,t)S_0(\vec{r},E) >}{< W(\vec{r},E), \mathbf{P}(\vec{r},E,t)S(\vec{r},E,t) >} \quad (15b)$$

$$\Delta\varrho_{mt}(t) \equiv \frac{< W(\vec{r},E), \Delta\mathbf{T}_{mt}(\vec{r},E,t)S_0(\vec{r},E) >}{< W(\vec{r},E), \mathbf{P}(\vec{r},E,t)S(\vec{r},E,t) >} \quad (15c)$$

$$\Delta\varrho_{md}(t) \equiv \frac{< W(\vec{r},E), \Delta\mathbf{T}_{md}(\vec{r},E,t)S_0(\vec{r},E) >}{< W(\vec{r},E), \mathbf{P}(\vec{r},E,t)S(\vec{r},E,t) >} \quad (15d)$$

$$\Delta\varrho_{other}(t) \equiv \frac{< W(\vec{r},E), \Delta\mathbf{T}_{other}(\vec{r},E,t)S_0(\vec{r},E) >}{< W(\vec{r},E), \mathbf{P}(\vec{r},E,t)S(\vec{r},E,t) >} \quad (15e)$$

such that

$$\Delta\varrho_{\Delta T}(t) = \Delta\varrho_{ppm}(t) + \Delta\varrho_{ft}(t) + \Delta\varrho_{mt}(t) + \Delta\varrho_{md}(t) + \Delta\varrho_{other}(t) \quad (16)$$

Similarly, using these contributions in (9c) the second–order terms are defined as

$$\Delta\varrho_{effect\Delta\phi}(t) \equiv \frac{< W(\vec{r},E), \Delta\mathbf{T}_{effect}(\vec{r},E,t)\Delta S(\vec{r},E,t) >}{< W(\vec{r},E), \mathbf{P}(\vec{r},E,t)S(\vec{r},E,t) >}, \quad \text{effect} \in \begin{Bmatrix} \text{ppm} \\ \text{ft} \\ \text{mt} \\ \text{md} \\ \text{other} \end{Bmatrix} \quad (17)$$

such that

$$\Delta\varrho_{\Delta T\Delta\phi}(t) = \Delta\varrho_{ppm\Delta\phi}(t) + \Delta\varrho_{ft\Delta\phi}(t) + \Delta\varrho_{mt\Delta\phi}(t) + \Delta\varrho_{md\Delta\phi}(t) + \Delta\varrho_{other\Delta\phi}(t) \quad (18)$$

The reactivity shift term $\Delta\varrho_\gamma$ is not easily separable because of its nonlinear terms in ΔP. However, if the fission cross sections do not change much during a transient, this term may be negligible compared to the others.

3. IMPLEMENTATION INTO PANBOX 2

In PANBOX 2, equation (4) is solv ith the nodal expansion method (NEM). Under the NEM discretization, auxiliary variables are defined for the neutron leakage $\mathbf{L}\phi$. Equation (4) is then discretized to the form of the 'nodal balance equation' or 'zeroth moment equation' which has the form

$$\left[(1 - \varrho_0)\mathbf{P}_0^m - \mathbf{A}_0^m\right]\vec{\phi}_0^m - \vec{J}_{net,0}^m = 0 \tag{19}$$

Here, the subscript m denotes the node number, and the vectors denote a vector of multigroup fluxes and net leakages. J_{net} is determined with the NEM outgoing current and auxiliary moment equations, as described in references 6 and 7. What is important for our purposes is that under the NEM discretization, the **L** operator does not appear explicitly in equation (19). While we could have chosen to involve the other NEM equations in our analysis – in order to isolate the reactivity contributions from changes in the L operator – we have instead chosen to lump these different effects together. Thus, we have defined the following reactivity contributions for our implementation in PANBOX 2.

$$\delta\varrho_{\Delta\phi,\gamma,J}(t) \equiv \frac{< W, [\mathbf{P}_0 - \mathbf{A}_0]\phi(t) - J_{out}(t) >}{< W, \mathbf{P}(t)\phi(t) >} - \frac{< W, [\mathbf{P}_0 - \mathbf{A}_0]\phi_0 - J_{out,0} >}{< W, \mathbf{P}_0\phi_0 >} \tag{20a}$$

$$= \Delta\varrho_{\Delta\phi} + \Delta\varrho_{\gamma} - \frac{< W, \Delta\mathbf{L}(t)S_0 >}{< W, \mathbf{P}(t)S(t) >} - \frac{< W, \Delta\mathbf{L}(t)\Delta S(t) >}{< W, \mathbf{P}(t)S(t) >}$$

$$\left.\begin{aligned} \delta\varrho_{effect}(t) &\equiv \frac{< W, [\Delta\mathbf{P}_{effect}(t) - \Delta\mathbf{A}_{effect}(t)]S_0 >}{< W, \mathbf{P}(t)S(t) >} && (20b) \\ \delta\varrho_{effect\Delta\phi}(t) &\equiv \frac{< W, [\Delta\mathbf{P}_{effect}(t) - \Delta\mathbf{A}_{effect}(t)]\Delta S(t) >}{< W, \mathbf{P}(t)S(t) >} && (20c) \end{aligned}\right\}, \quad \text{effect} \in \begin{Bmatrix} \text{ppm} \\ \text{ft} \\ \text{mt} \\ \text{md} \\ \text{other} \end{Bmatrix}$$

Thus, the entire change in reactivity since the beginning of the transient is expressed as a sum of all the components:

$$\begin{aligned} \Delta\varrho(t) = \delta\varrho_{\Delta\phi,\gamma,J} &+ \delta\varrho_{ppm} + \delta\varrho_{ft} + \delta\varrho_{mt} + \delta\varrho_{md} + \delta\varrho_{other} \\ &+ \delta\varrho_{ppm\Delta\phi} + \delta\varrho_{ft\Delta\phi} + \delta\varrho_{mt\Delta\phi} + \delta\varrho_{md\Delta\phi} + \delta\varrho_{other\Delta\phi} \end{aligned} \tag{21}$$

In the next section, we will give an example of how these reactivity components may be used to explain core phenomena during a boron dilution transient.

4. EXAMPLE CALCULATION

To demonstrate our method, we calculate a hypothetical boron dilution transient with the coupled RELAP5/PANBOX 2 system.The boron dilution is assumed to occur during natural circulation conditions when the reactor is highly subcritical and only producing power from the decay heat of fission products. During this transient, the boron concentration at the core inlet is steadily reduced until it reaches a level which will render the reactor prompt critical. The relative power and reactivity of the core is shown in Figure 1. Figure 2 shows the important contributions to the change in reactivity over time. As expected, when the boron dilution begins, the largest contribution to reactivity is due to changes in boron concentration. When the reactor reaches a prompt critical state, the strong power surge gives rise to an increase in the fuel temperature and a reduction in the moderator density. The feedback effects are seen in the strong negative reactivity components during this time period. Of great interest are the reactivity contributions between times t_1 and t_2. We have plotted the contributions along with the total reactivity in Figure 3. The contributions are translated to zero at the time when the reactor first becomes critical so that the phenomena are more clearly examined.

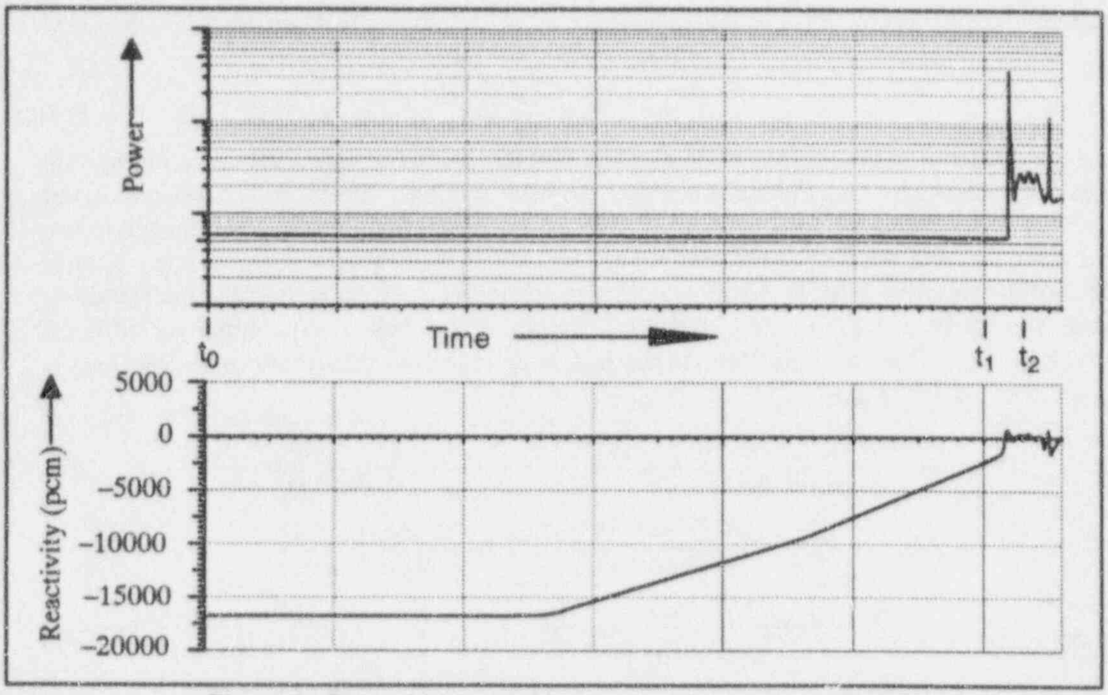


Figure 1: Power history and input reactivity of example transient.

From Figure 3, it is possible to study which feedback mechanisms are most important in abating the power surge. As the time of maximum reactivity is approached (t_3), increases in fuel temperature decrease the core reactivity. There is also some reduction of reactivity due to changes in the moderator density, however this effect is somewhat delayed until the energy generated during the surge is transferred from the fuel to the moderator. Most surprising is the reactivity contribution due to boron between times t_3 and t_4. Despite the fact that the boron concentration in the core is decreasing, the reactivity contribution of boron also decreases.

Figure 4 shows the separate contributions of $\delta\varrho_{ppm}$, $\delta\varrho_{ppm\Delta\phi}$ translated to zero from the time of maximum reactivity (t_3). It is seen from Figure 4 that the reduction in reactivity contribution due to boron comes from the second order term $\delta\varrho_{ppm\Delta\phi}$, which is the reactivity contribution of the change in boron concentration combined with the change in flux shape. Why this reactivity contribution is negative can be seen in Figure 5, where the core axial boron distribution and axial flux shape are plotted at times t_3 and t_4. The flux distribution in the core changes such that the net neutron absorption due to boron *increases*. It should be noted that this effect could not be calculated if first order perturbation theory had been used to calculate the reactivities for a point neutron kinetics model, since $\delta\varrho_{ppm\Delta\phi}$ would be neglected in that case. This point demonstrates the validity of the industry practice of using stationary three-dimensional flux calculations for determining point kinetics parameters, and furthermore highlights the usefulness of multidimensional kinetics analysis for such transients. Indeed, it is seen from these results that at least a one-dimensional neutron kinetics model is necessary to properly model the transient. Three-dimensional kinetics calculations would be needed if the boron dilution occurred only in one part of the core.

5. CONCLUSIONS

The simple reactivity edit option which we have developed for the coupled RELAP5/PANBOX 2 code system helps the analyst explain transient core physics phenomena. The abatement of a power surge during a hypothetical boron dilution transient can thus be fully explained with the calculation of reactivity components. In the case which we calculated, the change in the neutron flux shape proved to be a strong abatement factor of the power surge. Thus, it is very important that a multidimensional neutron kinetics model be used to analyze such transients.

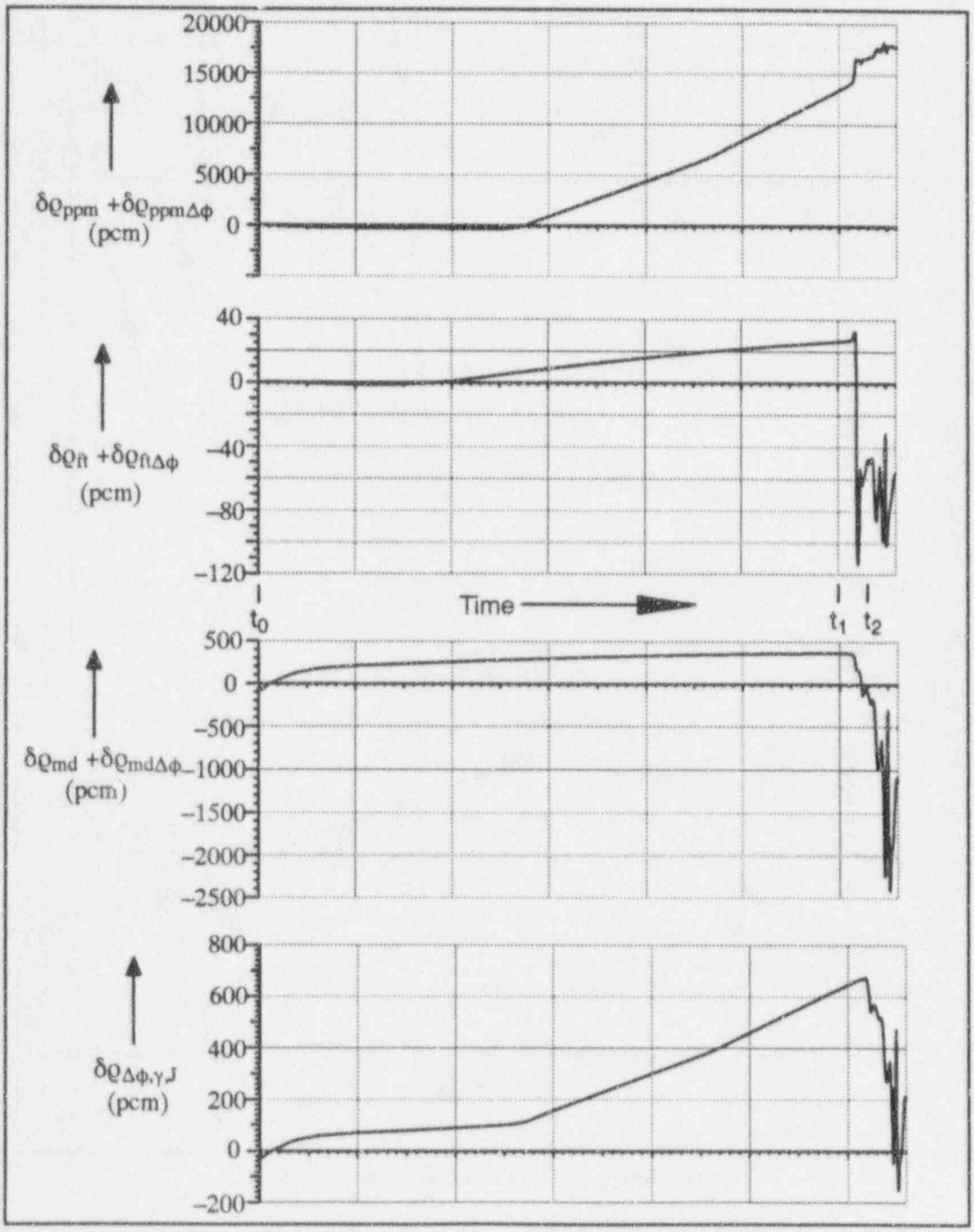


Figure 2: Important reactivity components of the transient.

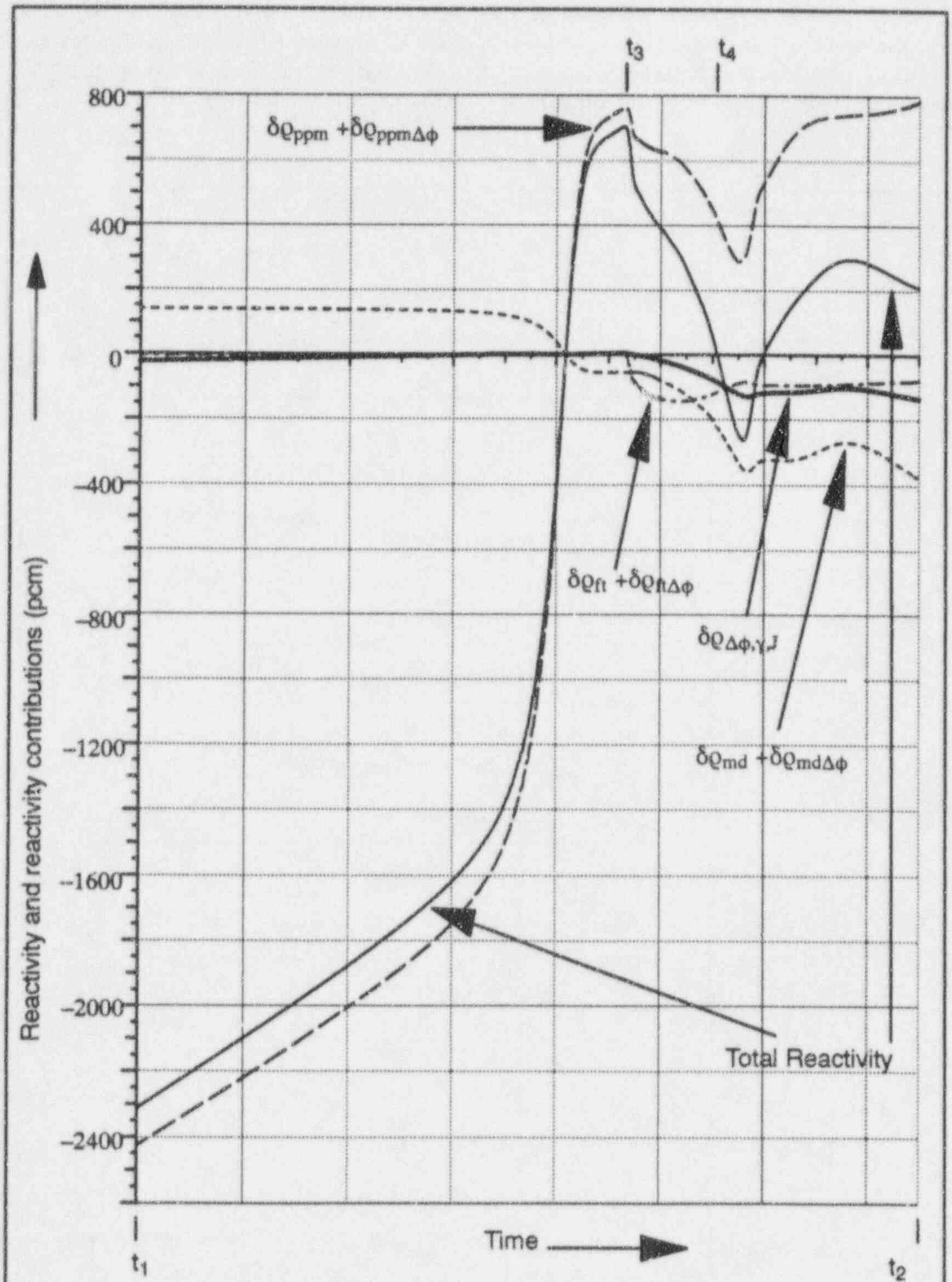


Figure 3: Total reactivity and reactivity components.

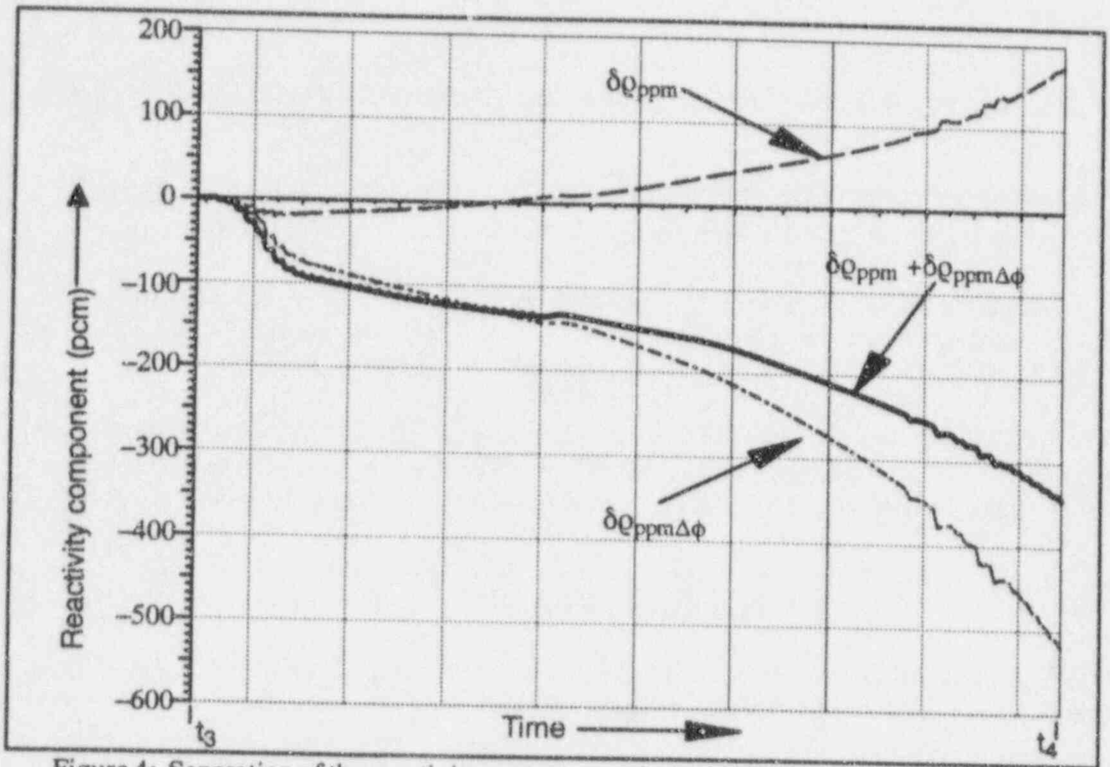


Figure 4: Separation of the reactivity components due to changes in boron concentration.

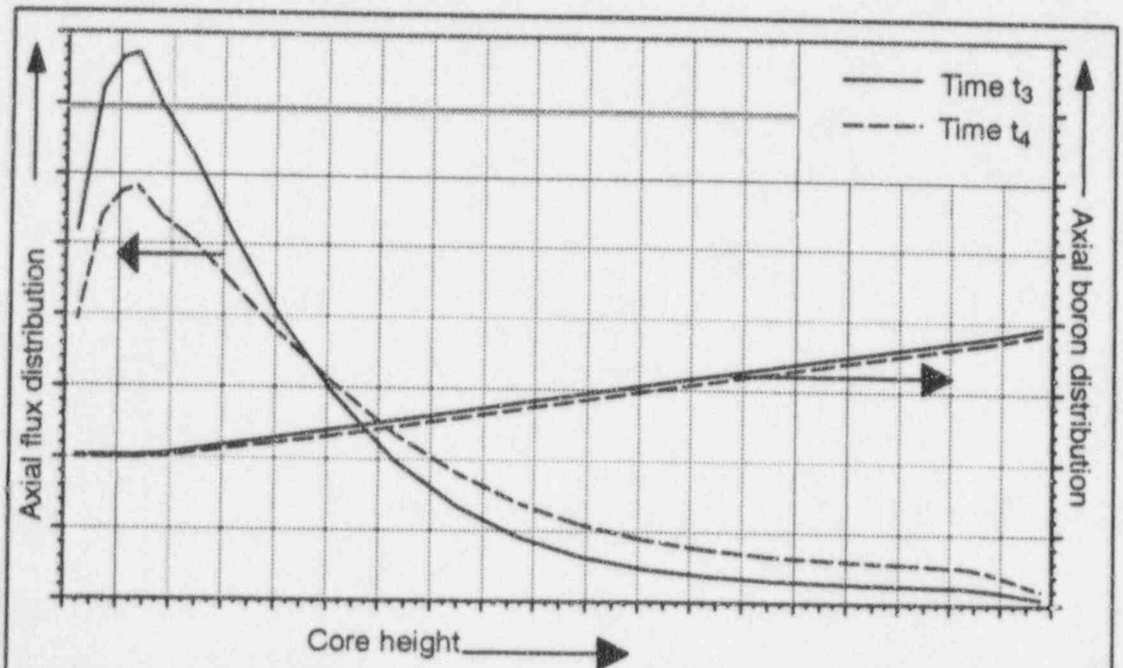


Figure 5: Axial flux and axial boron distributions at time of maximum reactivity and time of recriticality.

REFERENCES

(1) V. H. Ransom et al., *RELAP5/ MOD2 Code Manual,* NUREG/CR–4312, EGG–2396, Rev. 1, EG&G Idaho, Inc. (Mar. 1987).

(2) R. Böer, R. Böhm, H. Finnemann, R. Müller, 'The coupled neutronics and thermal hydraulics code system PANBOX for PWR safety analysis.' *Kerntechnik* 57 (1992) No. 1, pp. 49–54.

(3) A. Knoll, R. Müller, 'Coupling RELAP5 and PANBOX 2: A Three–Dimensional Space–Time Kinetics Application with RELAP5.' *International Conference on New Trends in Nuclear System Thermalhydraulics*, Pisa, Italy (May 30–June 2 1994).

(4) C. J. Jackson, H. Finnemann, 'Verification of the Coupled RELAP5/PANBOX2 System with the NEACRP LWR Core Transient Benchmark.' *International Conference on Mathematics and Computations, Reactor Physics, and Environmental Analyses*, Portland, Oregon, (April 30–May 4, 1995).

(5) Allan Henry, Nuclear Reactor Analysis, MIT Press, Cambridge, Massachusets (1975).

(6) H. Finnemann, F. Bennewitz, M.R. Wagner, 'Interface Current Techniques for Multidimensional Reactor Calculations.' *Atomkernenergie* 30 (1977) pp. 123–128.

(7) H.D. Fischer, H. Finnemann: 'The nodal integration method – a diverse solver for neutron diffusion problems.' *Atomkernenergie–Kerntechnik* 39 (1981) pp. 229–236.

ANALYSIS OF REACTIVITY TRANSIENTS USING A COUPLED THREE-DIMENSIONAL THERMAL-HYDRAULIC KINETICS CODE

K. Ivanov[1*], R. Macian[1], A. Irani[2], A. Baratta[1]

1) Department of Nuclear Engineering, The Pennsylvania State University, University Park, PA 16802

2) GPU Nuclear Corporation, One Upper Pond Road, Parsippany, NJ 07054

ABSTRACT

This paper summarizes the current status of Penn State's version of the coupled three-dimensional (3-D) thermal-hydraulic/kinetics TRAC-PF1/NEM code for PWR transient and accident analysis and describes future developments and applications.

The TRAC-PF1/NEM methodology utilizes closely coupled 3-D thermal-hydraulics and 3-D core neutronics transient models to simulate the vessel and a 1-D simulation of the primary system. This approach enables one to model in a "best estimate" manner complex transients involving 3-D effects. An efficient and flexible cross-section generation procedure was recently developed and implemented into TRAC-PF1/NEM. These features make the coupled code capable of modeling PWR reactivity transients, including boron dilution transients, in a reasonable amount of computer time. Three-dimensional studies on hot zero power (HZP) rod ejection and main steam line break (MSLB) transients in a PWR, as well as a LBLOCA accident, have been accomplished using TRAC-PF1/NEM. The results obtained demonstrate that this code is appropriate for the analysis of the space-dependent neutronics and thermal-hydraulic coupled phenomena related to many current safety issues.

1. Introduction

Reactivity transients in PWR's have been identified as a significant safety concern. Realistic numerical simulations of such transients require detailed 3-D thermal-hydraulic and more rigorous 3-D core dynamic models to better describe the space-time effects involved. Recent advances in computer technology make possible the incorporation of full 3-D modeling of reactor core into a system transient code. The application of these complex integrated safety codes to reactivity transient and accident analyses using the actual Nuclear Power Plant (NPP) operational data also requires an efficient cross-section generation methodology. The methodology must be capable of covering the whole range of core conditions experienced during different transients.

---

*Current address: Research Center Rossendorf Inc., Institute for Safety Research, P.O. Box 510119, D-01314 Dresden (on leave from the Institute of Nuclear Research and Nuclear Energy, Sofia).

To address these issues a closely coupled transient 3-D neutronic (NEM) /thermal-hydraulic PWR analysis code using TRAC-PF1 has been developed at the Pennsylvania State University (PSU). Although further improvements are needed for TRAC-PF1/NEM, this code has reached the level suitable for direct industry application.

2. Coupled TRAC-PF1/NEM Methodology

Among current best estimate system transient codes TRAC-PF1 has a three-dimensional thermal-hydraulic analysis capability[1]. A modified version of TRAC-PF1/MOD2 v.5.4 is currently being used at PSU. This version incorporates a 1-D decay heat model, that dynamically computes the decay heat axial shape during the transient. The code calculates general transient two-phase coolant conditions in one, two, or three dimensions using a realistic six-equation, two-fluid, finite-difference model. This six-equation model, in conjunction with specialized empirical models for a variety of PWR primary- and secondary-loop components and control systems, allows TRAC-PF1/MOD2 v.5.4 to accurately model both mild and severe thermal-hydraulic transients.

An accurate 3-D transient neutronics model based on the Nodal Expansion Method (NEM) has been developed and integrated into TRAC-PF1[2]. The NEM spatial model is based on the transverse integrated procedure. Two levels of approximation have been used: fourth-degree transverse-integrated flux representation and quadratic leakage approximation. The nodal coupling relationships are expressed in a partial current formulation. The time dependence of the neutron flux is approximated by a first order, fully implicit, finite-difference scheme, whereas the time dependence of the neutron precursor distributions is modeled by a linear time-integrated approximation. The coarse-mesh rebalance and asymptotic extrapolation methods are used to accelerate convergence of the iterative solution process. Several benchmark problems have been used to assess the NEM model in both steady-state and transient conditions[2,3]. Very good agreement was obtained among the reference results and those from NEM.

The coupling of the NEM neutronics to TRAC-PF1 has made use of the local thermal-hydraulic description to simulate the core response during a transient. TRAC-PF1/NEM employs an improved semi-implicit neutronics/thermal-hydraulic coupling scheme. At the beginning of a time step, TRAC-PF1/MOD2 first performs its prepass stage, where fluid-state-dependent material properties and heat-transfer coefficients are calculated based on thermal-hydraulic conditions at the end of the previous time step. Then in the outer iteration stage, the multidimensional fluid-dynamic equations are solved using previous time-step fuel-rod heat fluxes. Then, the 3-D transient NEM neutronics model calculates the present time-step nodal power distribution using cross-section-dependent feedback parameters based on present time-step fluid conditions and previous time-step fuel-rod temperatures. Finally, in the postpass stage, the new nodal power distribution is used in the numerical solution of the heat-conduction equations. To ensure symmetry between the 3-D thermal-hydraulics vessel, the heat structure, and the neutronics core model proper radial and axial noding and mapping schemes have to be developed for a given reactor and for a given transient.

The TRAC-PF1/NEM cross-section modeling algorithm has two options implemented. These two options are based on different cross-section generation procedures. The first option uses the standard polynomial fitting procedure and is designed mainly for carrying out benchmark calculations.This model has, however, revealed some disadvantages in real reactivity transient and accident simulations, since it has proved to be accurate only for small perturbations. In case the system conditions during a transient simulation are beyond the range of validity of polynomial fits, there is a possibility of negative values for cross-sections. This technique may also lead to non-converging solutions in a highly heterogeneous core environment. In order to solve these problems, a cross-section dependency model, based on a linear interpolation in four-dimensional tables has been developed. Up to four independent thermal-hydraulic parameters that contribute to the neutronic feedback can be selected. In this way the neutronic calculations use a linear interpolation scheme with tables of cross-sectional data as a function of the chosen parameters. The expected range of the main thermal-hydraulic parameters for a given transient is covered by the selection of an adequate range for the independent variables. This approach allows a wider range of core conditions to be covered, keeps the cross-section values within the appropriate ranges and enhances the NEM convergence.

The correct reproduction of initial steady-state core conditions at given point in the cycle depletion is based on the output information provided by so-called core simulator codes (e.g. SIMULATE-3). The fuel assembly layout, fuel assembly axial compositions, as well as 2-D and 3-D exposure distributions are used to develop a 3-D NEM model by proper axial nodalization and cross-section mapping. The cross section set tables are generated using CASMO-3 (or other suitable fuel assembly burnup program). Two NEM cross-section libraries are built, one for fully rodded and the other for fully unrodded compositions. The cross-section library generation process is automated. The NEM core mapping process is also time consuming, especially in the determining the axial exposure approximation, and its automation is under development.

A general control rod presence and movement algorithm has been implemented in TRAC-PF1/NEM and integrated into the cross-section tables procedure. Cross sections for nodes with control rods partially inserted are obtained by blending the rodded and unrodded cross sections, using a factor that is a function of the fractional amount of control-rod insertion in that cell. The developed control-rod algorithm is capable of modeling initial steady-state conditions with the initial positions of the control-rod groups. Movement of single rods and control-rod groups, as well as dynamic scram can also be simulated. In addition to this flexibility, there is an option in the code that allows one to perform efficient evaluations of static control-rod reactivity and shutdown worths.

3. Application of TRAC-PF1/NEM to Reactivity Transient Analyses

Using the above-described methodology, three-dimensional simulations of a rapid off-center control-rod ejection in a Westinghouse four-loop PWR at HZP[3,4], of the post-LBLOCA reflood phase in a PWR[5], and of a MSLB scenario for hot full power (HFP) conditions at the end of cycle (EOC) (TMI-1 NPP, Cycle 10)[6] have been performed. These studies have been accomplished using an existing network of workstations operated by PSU. In this paper selected results from two of these applications are presented and discussed in order to demonstrate the capabilities and performance of the coupled code, as well as the functionality of different code options.

The HZP rod ejection transient provides a stringent test of the NEM transient routine and its coupling to the fuel-rod heat-conduction solution methodology. Because actual reactor-core temperatures and power distributions are difficult to measure accurately under such severe transient conditions, the results from running TRAC-PF1/NEM have been compared with the results of running two established neutronics/thermal-hydraulic space-time codes HERMITE and ARROTTA. Steady-state and transient calculations with one and four radial nodes per assembly (npa) NEM models have been performed. Comparisons of steady-state calculations of initial and asymptotic conditions showed very good agreement in the calculated eigenvalues (k_{eff}) and normalized assembly power (NP) distributions. The steady state results obtained by the 4 npa NEM model and comparisons with 4 npa HERMITE and 1 npa ARROTTA calculations are summarized in Table 1. Good agreement has been obtained also among the predictions from TRAC-PF1/NEM (with the 4 npa NEM model and a fixed time step size of 1 ms), HERMITE, and ARROTA during all phases of this transient. Agreement was in the areas of time dependence of total core power and peak-assembly power density, as well as the time dependence of the core-average and peak-assembly fuel temperatures. The time behavior of the above-mentioned quantities during the transient out to 0.5 s is shown in Figs. 1-4. The normalized assembly power distribution at 0.39 s into the transient (near the power peak of the transient) has been studied and compared also. The ARROTTA normalized assembly powers agree to within 5.02% with those of HERMITE while the TRAC-PF1/NEM normalized assembly powers agree within 0.55% with those of HERMITE. In conclusion, this PWR HZP rod ejection simulation indicated that the coupled TRAC-PF1/NEM code produces results within the range of results obtained by HERMITE and ARROTTA.

To further test the coolant thermal-hydraulic coupling a MSLB transient study has been accomplished using TRAC-PF1/NEM. The MSLB accident in a PWR is characterized by significant space-time effects in the core caused by asymmetric cooling and assumed stuck-out rod during reactor trip. This MSLB analysis assumed the instantaneous quillotine break of one of the two main steam lines upstream of the main steam isolation valve. The reactor is at EOC when the moderator temperature coefficient (MTC) is most negative. The concurrent failure of a feedwater regulating valve to the affected steam generator is also assumed. A reactor scram occurs on high neutron power or low system pressure. The scram time of 2.2 s for 100% insertion of the control-rod groups is used. The most reactive rod is assumed to be stuck out. Feedwater (FW) to the affected steam generator is terminated by closure of the FW block valve at 55 s (1-st scenario) or at 105 s (2-nd scenario) into the transient. High pressure injection is assumed not to activate.

The TRAC-PF1/NEM results for initial steady state conditions have been compared with the SIMULATE-3 results and the agreement is acceptable.

Comparisons of NP distributions and axial exposure and power shapes (for one representative assembly position) are shown in Figs. 5 through 7. Figs. 8 through 10 display 2-D NP maps of the core at 0 s, 8.5 s and 100 s into the MSLB transient. Fig. 8 shows the initial insertion of Group 7 while Fig. 10 demonstrates the flux redistribution at the location of the stuck rod. During the transient a comparison of the 3-D kinetics results to a compatible point kinetics prediction has been performed. We see in Figs. 11 and 12 the total power behavior for both FW scenarios predicted by the point kinetics and 3-D kinetics models. These figures show that TRAC-PF1 with point kinetics predicts a return to power, with a maximum power level dependent on the closure time of FW, while TRAC-PF1/NEM does not. TRAC-PF1/NEM incorporates 3-D transient local nodal power information based on local thermal-hydraulic feedback, whereas in the point kinetics model, only entire core average conditions are used. This study demonstrates that the 3-D core transient modeling provides margin to re-criticality over a point kinetics approach during a MSLB analysis.

A new model for best estimate modeling of boron mixing and transport is being developed and implemented in TRAC-PF1 at PSU[7]. The coupled TRAC-PF1/NEM methodology, utilizing this high order boron tracking algorithm, will be applied to the study of boron dilution transients in PWR's.

4. Conclusions

Additional improvements are planned for introduction into TRAC-PF1/NEM. Assembly discontinuity factors are being utilized in both the basic NEM algorithm and cross-section tables procedure in order to enhance the accuracy of the neutronic simulation in real applications. The existing control-rod modeling algorithm needs to be modified to account for rod cusping effects. Studies on the numerical convergence of coupled neutronics/thermal-hydraulic calculations have been initiated and further effort has to be directed toward the development of numerical techniques that more closely approximate a truly coupled solution.

Results from several PWR studies show a good accuracy in both steady state power distribution predictions and transient power evolution. The coupled TRAC-PF1/NEM code performs well during simulations where different feedback effects are significant. TRAC-PF1 has a full 3-D thermal-hydraulic description of the vessel flow and is, therefore, able to resolve local effects that would be impossible to study with less sophisticated codes based on a parallel coolant channel model. The coupled TRAC-PF1 /NEM methodology combines the 3-D thermal-hydraulic models of TRAC-PF1 with a fully transient 3-D neutronics simulation of the core. The code is supplemented by an efficient and flexible cross-section generation procedure. The calculations performed demonstrate that TRAC-PF1/NEM resolves local and system interaction effects in a reasonable amount of computational time. All these features make the PSU version of TRAC-PF1/NEM capable of modeling PWR reactivity transients where an accurate 3-D simulation is required.

References

1. N. M. Schnurr at al., "TRAC-PF1/MOD2: Volume1. Theory Manual," Los Alamos National Laboratory report LA-12031-M, Vol. I (NUREG/CR-5673) (1992).

2. B. R. Bandini, "A Three Dimensional Transient Neutronics Routine for the TRAC-PF1 Reactor Thermal Hydraulic Computer Code," Ph.D. thesis, The Pennsylvania State University (1990).

3. B. R. Bandini, A. J. Baratta, and R. J. Steinke, "Verification of a Three-Dimensional Nodal Transient Neutronics Routine for the TRAC-PF1/MOD3 Thermal Hydraulic System Analysis Code," LA-UR-92-1210, Los Alamos National Laboratory (1992).

4. K. N. Ivanov, A. J. Baratta, and B. R. Bandini, "Three-Dimensional NEM/TRAC-PF1 Rod Ejection Benchmark Analyses," *Trans. Amer. Nuc. Soc.* Vol. 72, p. 343 (1995).

5. T. N. Tyler, R. Macian, and J. H. Mahaffy, "Analysis of the Return to Power Scenario Following a LBLOCA in a PWR," *Proc. of NURETH-7, 2669-2676,* Saratoga Springs (1995).

6. K. N. Ivanov, R. Macian, A. J. Baratta, A. Irani, J. W. Folsom, and N. G. Trikouros, "TMI MSLB Analysis Using TRAC-PF1 Coupled with Three-Dimensional Kinetics," accepted for presentation at the 1995 ANS Winter Meeting, San Francisko, October 29 - November 1 (1995).

7. R. Macian and J. H. Mahaffy, Department of Nuclear Engineering, The Pennsylvania State University, Personal Communication (September 1995).

Code	Keff I.C.[a]	Keff A.C.[b]	Ejected Rod Worth (Absolute)	NPmax I.C.	NPmax A.C.
HERMITE	0.986523	0.995033	0.008510	1.8116	5.0832
ARROTTA	0.987026	0.995479	0.008453	1.8544	5.1017
% DIFF	0.050988	0.044819	-0.6740	2.36	0.36
TRAC-PF1 /NEM	0.986499	0.994999	0.008500	1.8091	5.0693
% DIFF	-0.002429	-0.003420	-0.1175	-0.14	-0.27

[a]Initial Conditions; [b]Asymptotic Conditions

Table 1. Comparison of TRAC-PF1/NEM, ARROTTA and HERMITE (as reference) Steady State Results for PWR HZP Rod Ejection Calculations.

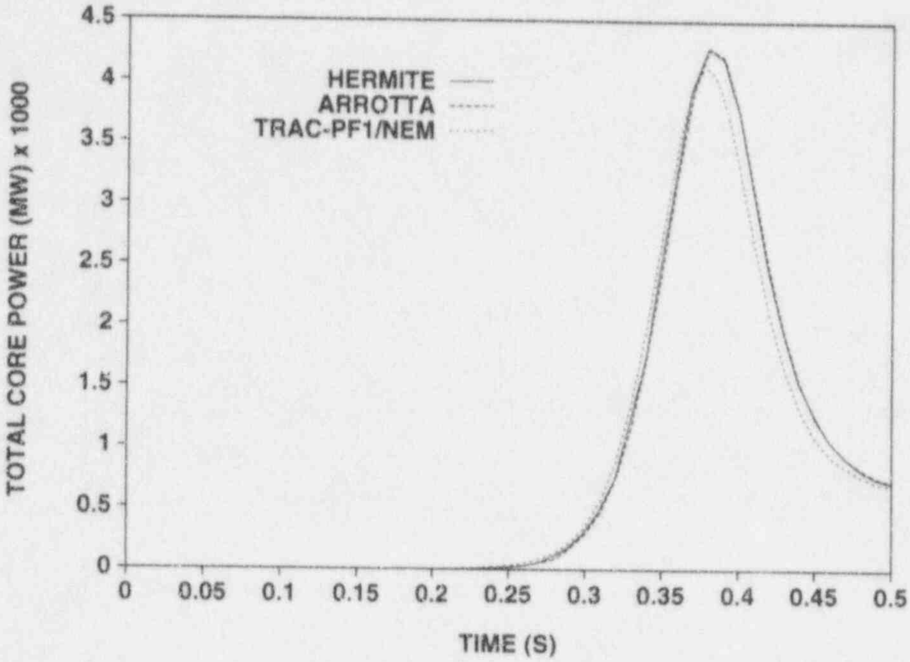


Fig. 1. Total Core Power as a Function of Time.

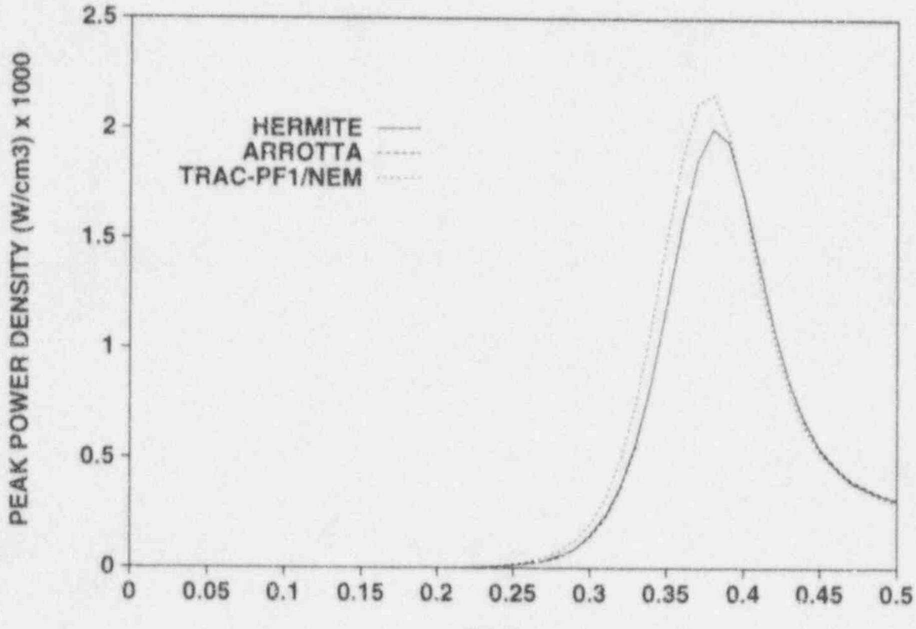


Fig. 2. Peak Assembly Power Density as a Function of Time.

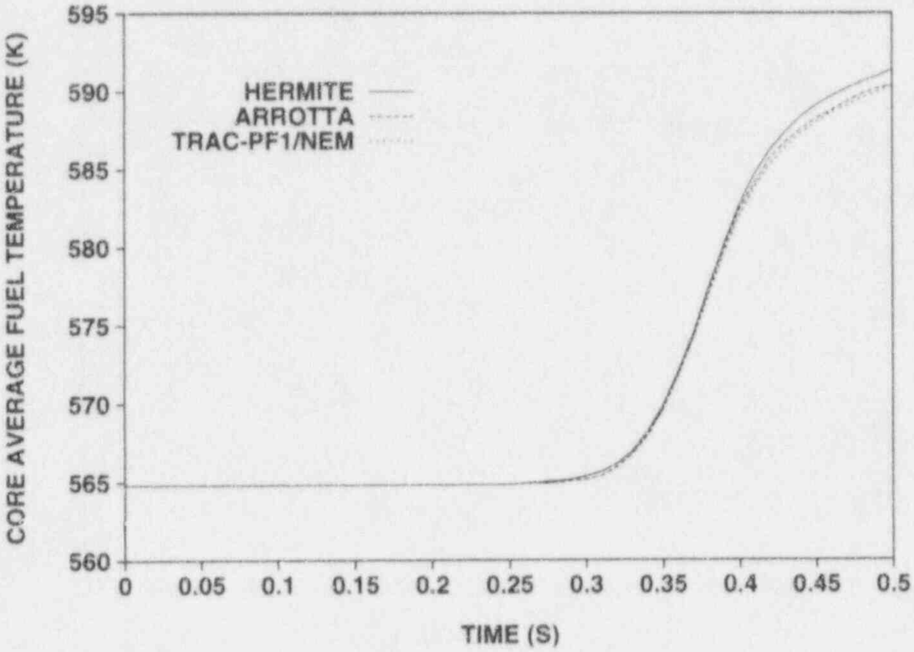


Fig. 3. Core Average Fuel Temperature as a Function of Time.

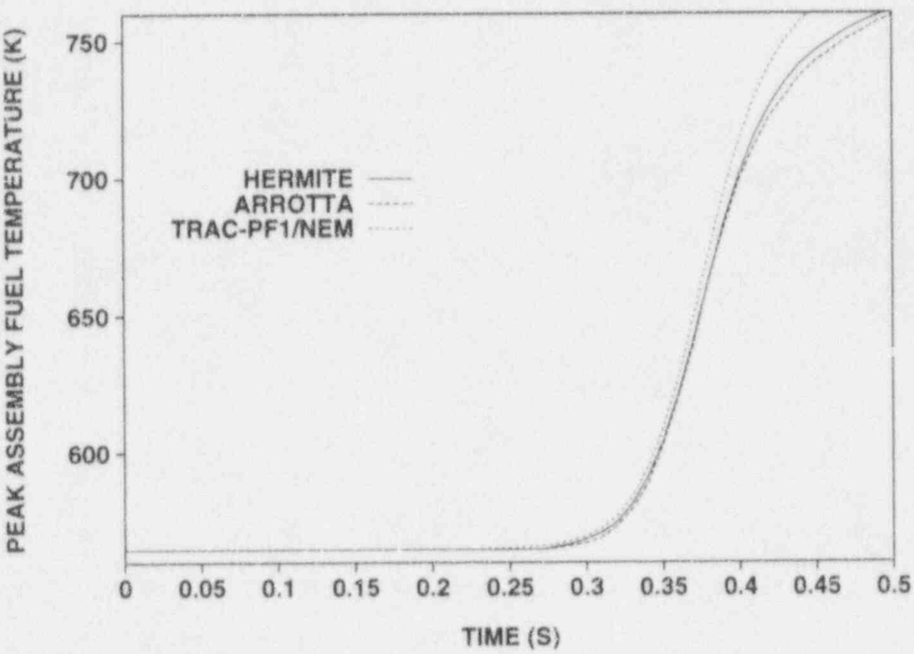


Fig. 4. Peak Assembly Fuel Temperature as a Function of Time.

	08	09	10	11	12	13	14	15
H	0.965 +0.005	1.316 -0.042	1.029 +0.006	1.058 -0.048	1.069 -0.023	1.299 -0.044	0.931 -0.055	0.389 +0.002
K		1.022 +0.007	1.309 -0.020	1.045 +0.008	1.309 -0.022	1.086 -0.005	1.130 -0.045	0.393 +0.007
L			1.066 +0.024	1.320 +0.011	1.022 +0.012	1.298 -0.011	1.065 -0.026	0.343 +0.026
M				1.061 +0.021	1.298 -0.017	0.980 +0.012	0.973 +0.035	
N					1.015 +0.008	1.056 -0.024	0.477 +0.024	
O						0.546 +0.023	A B	
P								
R								

A - SIMULATE-3 Reference (4 npa Model with ADFs)

B - NEM/TRAC (1 npa Model without ADFs) - SIMULATE-3

Normalized Assembly Power Distribution at EOC

TMI, Unit 1, Cycle 10

Fig. 5. Two-Dimensional Power Map.

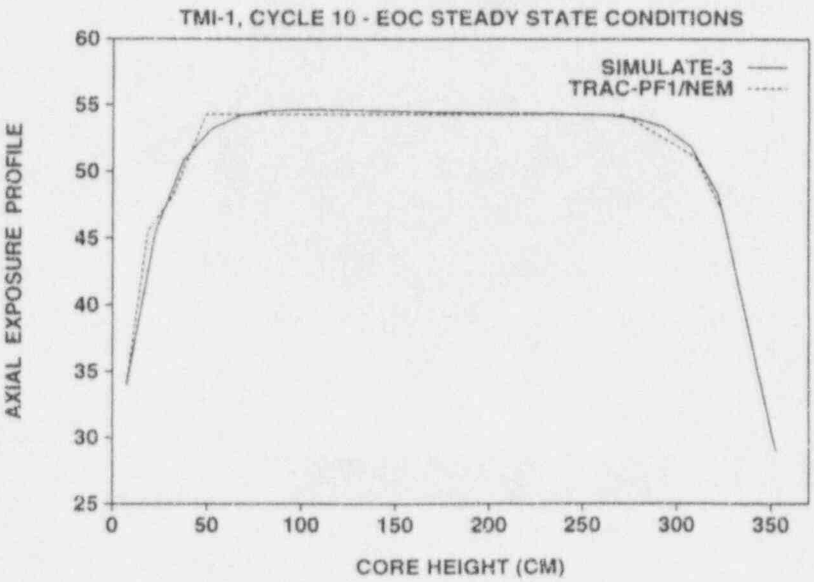


Fig. 6. Axial Exposure Distribution for Assembly at Postion H08.

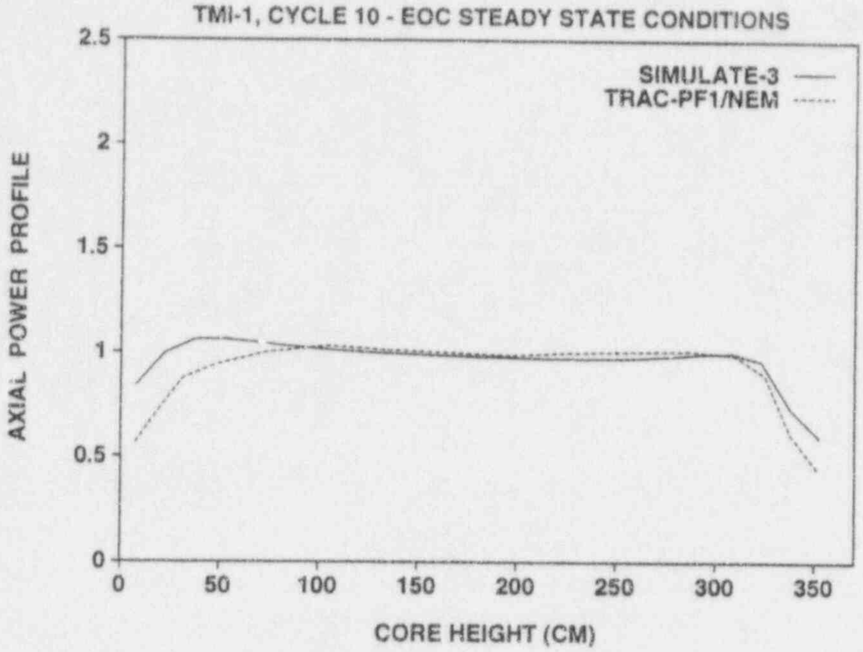


Fig. 7. Axial Power Distribution for Assembly at Position H08.

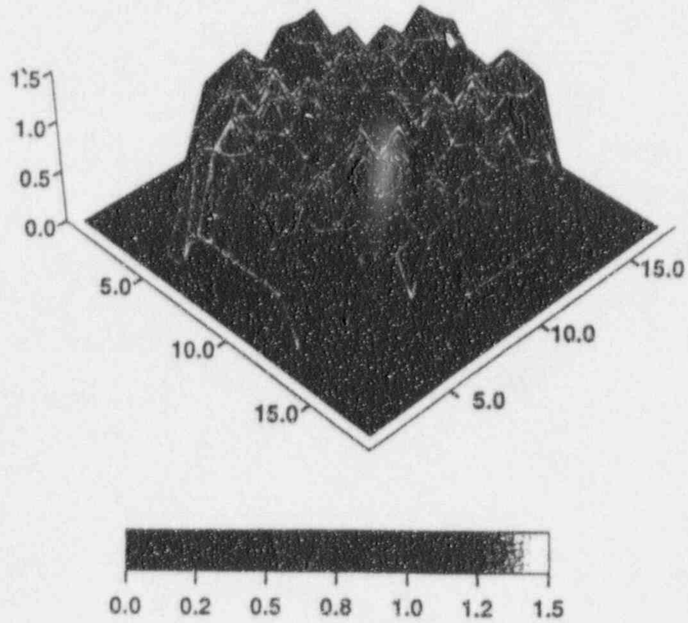


Fig. 8. Initial Steady State Normalized Assembly Power Distribution.

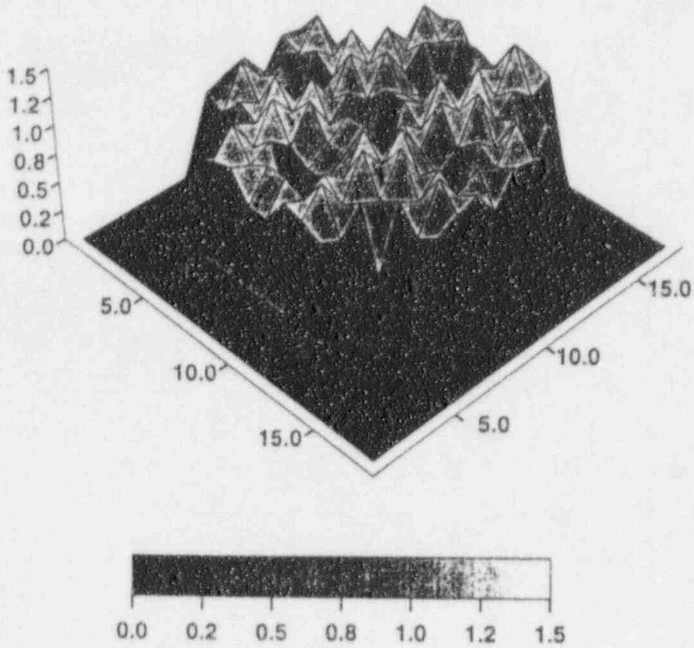


Fig. 9. Normalized Assembly Power Distribution at 8.5. s into MSLB - before Scram.

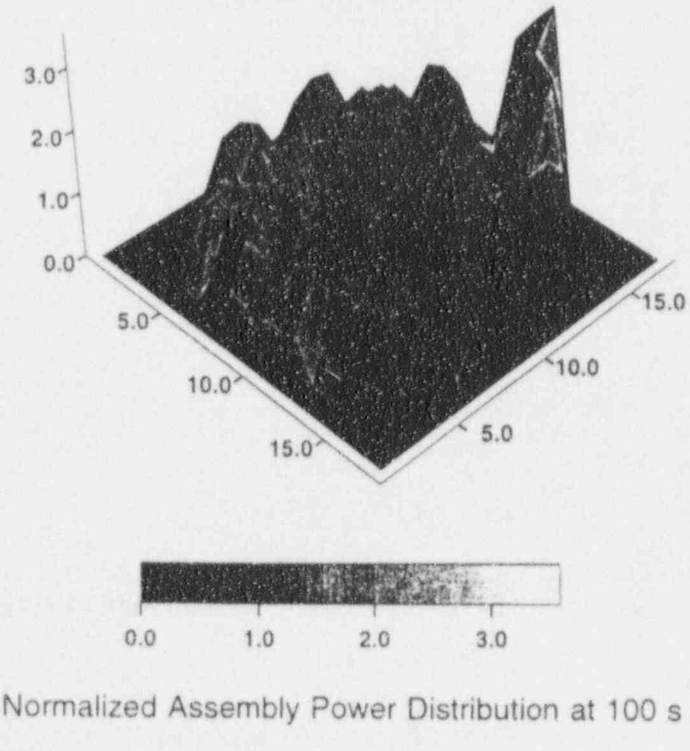


Fig. 10. Normalized Assembly Power Distribution at 100 s into MSLB.

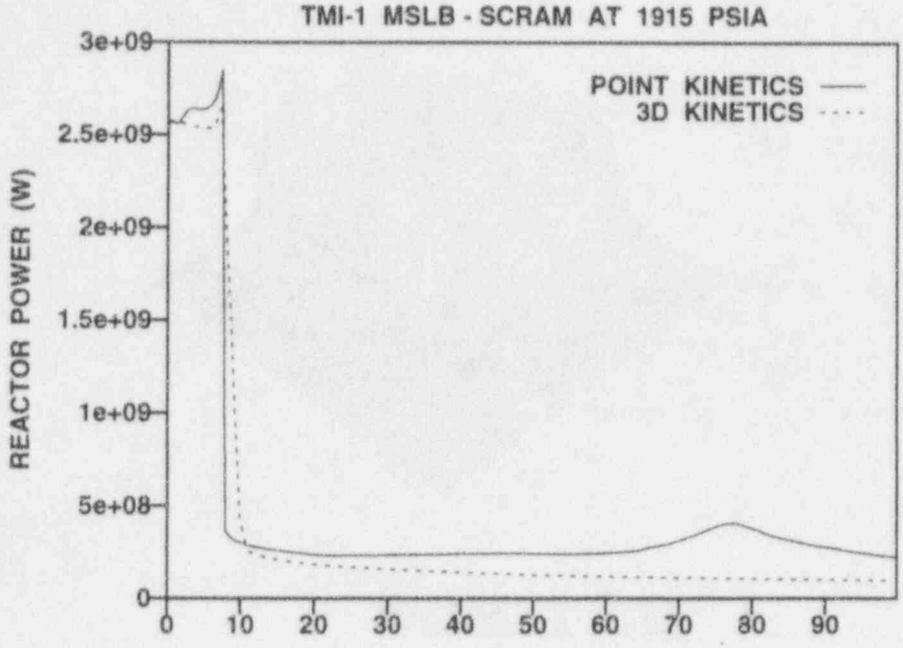


Fig. 11. Power Behavior - 55 s Feedwater.

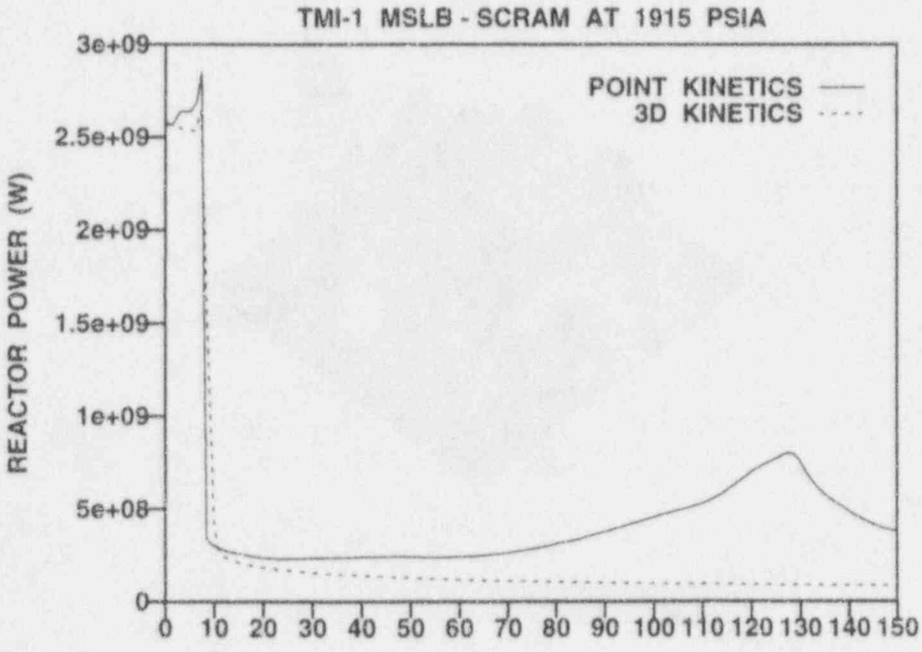


Fig. 12. Power Behavior - 105 s Feedwater.

OECD/NEA/CSNI Specialist Meeting on Boron Dilution Reactivity Transients
State College, Pennsylvania, U.S.A., October 18 - 20, 1995

BORON DILUTION TRANSIENT CALCULATION WITH THE HEXTRAN CODE USING CFDPLIM

Markku Rajamäki and Riitta Kyrki-Rajamäki
VTT Energy, Nuclear Energy
P.O.Box 1604, FIN-02044 VTT, Finland
Tel. +358 0 456 5016, Fax. +358 0 456 5000
E-mail: Markku.Rajamaki@vtt.fi

ABSTRACT

A main analysis tool in Finland for reactor dynamical RIA calculations has been the three-dimensional HEXTRAN code which also includes full circuit models. Reliable calculation of propagating boron fronts is very difficult with standard numerical algorithms because numerical diffusion tends to smoothen the front. In this way, the reactivity effect of the boron dilution can be significantly lowered and conservatism of the analyses cannot be guaranteed. In normal flow conditions this problem has been avoided in HEXTRAN analyses by simulating the dilution front directly to the core inlet. In natural circulation conditions there occurs significant numerical diffusion even during the propagation of boron front inside the core. The new hydraulics solution method PLIM (Piecewise Linear Interpolation Method) is capable to avoid the excessive errors, the numerical diffusion and also the numerical dispersion.Therefore a new hydraulics solver CFDPLIM using PLIM has been applied to HEXTRAN. Examples are given of comparative analyses made with HEXTRAN in both flow conditions. The results show excellent performance of the solver also in varying flow conditions when the rigid discretization is dictated by neutronics.

1. INTRODUCTION

Analyses of the possibilities and potential consequences of local boron dilution are going on for the VVER-440 reactors in Loviisa, Finland. Reactor dynamics calculations of local boron dilutions both in normal and in standby conditions and during accidents have been and are being made in VTT Energy in cooperation with IVO International Ltd[1] .

The reactor dynamics calculation system of VTT Energy covers both reactor types in Finland - the ABB Atom type BWRs in Olkiluoto and the VVER type PWRs in Loviisa. HEXTRAN is a three-dimensional hexagonal transient and accident analysis code developed in VTT for VVERs. The thermal hydraulics model SMABRE is dynamically coupled with HEXTRAN for the primary and secondary circuit calculations. It is possible to make with HEXTRAN fully realistic three-dimensional dynamical core analyses starting from accurate core conditions.

HEXTRAN has been thoroughly validated and extensively applied to safety analyses of the Loviisa power station, the new Russian concept VVER-91 and the Hungarian VVER-440 type NPP Paks. Main applications are the calculations of asymmetric accidents in the reactor core originating from neutronic or thermal hydraulic disturbances in core or cooling circuits. HEXTRAN needs only a few seconds of CPU time for calculation of one time-step with HP's 9000/715 UNIX workstations, so also such long accidents as Anticipated Transients Without Scram (ATWS) have been analyzed.

In the analyses of RIAs, the thermal hydraulics and reactor kinetics processes in the reactor core are coupled in such a manner that the equations for the processes have to be solved simultaneously by iteration and fully consistent solutions have to be achieved. The iteration is governed simplest when neutronics and thermal hydraulics use the same discretization. Neutronics is very sensitive to any small changes, also to those which happen in numerical errors. Therefore a varying discretization in hydraulics produces unreliable results. Reliable calculation of propagating boron fronts is very difficult with standard numerical algorithms because numerical diffusion tends to smoothen the front. In this way, the reactivity effect of the boron dilution can be significantly lowered and conservatism of the analyses cannot be guaranteed. Although the sufficient accuracy in the case of one propagating quantity can be achieved by careful choice of the discretization, the situation is different to several flow channels. There are several, varying, unknown velocities, which cause several, varying needs, and it is impossible to the discretization to satisfy all of them. Therefore, the flow equations of hydraulics should be possible to be solved accurately in any given discretization, now in that which is dictated mainly by neutronics.

A new solution method PLIM (Piecewise Linear Interpolation Method) has been developed at VTT for the system of time-dependent one-dimensional flow equations. PLIM is capable to avoid numerical diffusion and numerical dispersion in any discretization grid. The latest computer version CFDPLIM has been attached to HEXTRAN as a part of its hydraulic model.

In Chapter II the main properties of HEXTRAN are described. In Chapter III the properties of the new hydraulics solution method PLIM and the solver CFDPLIM applied in the new version of HEXTRAN (HEXTRAN-PLIM) are presented. In Chapter IV the effects of the new solution method on the boron dilution calculations are shown in different flow conditions. Conclusions are drawn in Chapter V.

2. MODELS OF HEXTRAN

HEXTRAN[2] is developed for analyses of VVER-440 and VVER-1000 reactors. It is based on the three-dimensional stationary fuel management code HEXBU-3D[3] and on the one-dimensional reactor dynamics code TRAB[4]. It describes accurately the VVER core consisting of hexagonal fuel elements. HEXTRAN is intended for calculation of accidents, where radially asymmetric phenomena are included and both neutron dynamics and two-phase thermal hydraulics are important, e.g. control rod ejection. The thermal hydraulic circuit model SMABRE[5] has been dynamically coupled to HEXTRAN for analyses of accidents including effects of the whole cooling system, such as main steam line break or startup of an inoperable loop. Different connection geometries between HEXTRAN and SMABRE can be defined.

The circuit hydraulics solution consists of five conservation equations for mass and enthalpy of vapor and liquid and the momentum for the mixture. The phase separation modelling is based on the drift-flux approach. The process description is based on generalized nodes, junctions connecting nodes and heat structures describing structure walls, fuel rods and steam generator tubes. Advanced fast non-iterative numerical schemes applying sparse matrix solvers are used for the solution of discretized conservation equations.

In the core models advanced time integration methods are used. Time discretization is made by implicit methods which allow flexible choices of time-steps. In the thermal hydraulics of the core the numerical method for conservation equations can be varied between central difference and fully implicit.

The neutron kinetics model of HEXTRAN solves the two-group diffusion equations in a homogenized fuel assembly geometry by a sophisticated, fast nodal method. Within nodes the two-group fluxes are represented by linear combinations of two time-dependent spatial modes, the fundamental and the transient mode of solution. The dynamic equations include normal six groups of delayed neutrons. The feedback effects from xenon-poisoning, fuel and coolant temperatures, coolant and soluble boron densities are included in the code. A special treatment has been developed for the dynamic calculation of the moving fuel assemblies of big follower-type control elements which are used in the VVER-440 type reactors. In addition to full core calculations, core symmetries for half core, 1/3 or 1/6 can be utilized. Burnup data evaluated with HEXBU-3D can be used as such in HEXTRAN and also the kinetic constants are burnup-dependent.

The thermal hydraulics in the core is calculated in separated axial hydraulic channels, which connect freely to one or several fuel assemblies. In VVER-440 type reactors there is almost no mixing between the hydraulic channels in the core because there are shroud tubes around individual fuel assemblies. Channel hydraulics is governed by the conservation equations for steam mass, water mass, total enthalpy and total momentum, and by appropriate correlations. The mass flow distribution between the channels is based on the pressure balance over the core. The phase velocities may be interconnected by the drift-flux formalism. The properties of water and steam are represented as rational functions of pressure and enthalpy.

In order to get an accurate representation of the fuel temperature based Doppler feedback the heat transfer calculation with several radial meshes is made for an average fuel rod in each fuel assembly (divided also axially in several nodes). The prompt and delayed release of nuclear heat in fuel or in coolant is also modelled. The heat conduction equation is solved according to Fourier's law with temperature dependent thermal properties of fuel pellet, gas gap and fuel cladding and with different heat transfer coefficients for different hydraulic regimes.

The hot channel calculations are made afterwards on the basis of the output files of HEXTRAN with the one-dimensional reactor dynamics code TRAB. The extreme phenomena modelled are the fuel temperature rise after occurrence of the boiling crisis and oxidation of the cladding material. Sensitivity studies with different conservatisms can easily be carried out.

There is not very much experience in the world in making conservative safety analyses with a best-estimate three-dimensional reactor dynamics code. A new multiple hot channel methodology has been developed for this purpose.

A summary report on the validation of HEXTRAN has been published in the STUK series[6]. HEXTRAN is based on already validated codes and the models of these codes have been shown to function correctly also within the HEXTRAN code. The main new model of HEXTRAN, the spatial kinetics model has been successfully validated against Czech LR-0 test reactor and Loviisa plant measurements. In the stationary state the neutronic solution of HEXTRAN is the same as that of HEXBU-3D, which has been validated by thorough comparisons with Loviisa plant data.

The reactor dynamics analyses are supported by the reactor physics calculation system of VTT Energy. Reactor physics data for the dynamics codes can be generated starting from the basic nuclear data libraries, by using VTT's hexagonal version of the CASMO code, CASMO-HEX.

3. THE HYDRAULICS SOLVER CFDPLIM

An essential problem in computational fluid dynamics is the avoidance of typical numerical errors. These numerical errors cause diffusion and oscillations in the most essential parts of the solution. A new shape-preserving characteristics method, Piecewise Linear Interpolation Method, PLIM[7, 8] has been recently developed at VTT for solving the system of time-dependent one-dimensional flow equations.

Originally PLIM was capable to handle the system of equations as

$$
\begin{aligned}
&\frac{\partial}{\partial t} U(u, z, t) + \frac{\partial}{\partial z} F(u, z, t) \\
&+ \sum_m [\, A_m(z,t) \frac{\partial}{\partial t} X_m(u, z, t) + B_m(z,t) \frac{\partial}{\partial z} Y_m(u, z, t) \,] \qquad (1) \\
&= P_S(u, z, t)
\end{aligned}
$$

where **u** is the unknown vector. It is implicitly assumed that the equations are physical, what means now that the equations form an initial value problem in respect of time.

The particular features of the PLIM algorithm can be described as follows

- PLIM divides the z-t-domain into mesh cells, which are treated in the algorithm separately.
- Within the mesh cell local linear approximations in respect of **u** are utilized for the terms of the equations.
- The system of the equations are transformed to the characteristic form.
- The unknown variables $x_i(z,t)$ of the characteristic equations are, firstly, treated indepently from each others.
- The solution to them are obtained according to the characteristic method where an essential step is to interpolate the value of $x_i(z,t)$ at the incoming boundary of the mesh cell.
- At the interfaces of the mesh cells PLIM forms a piecewice linear approximation using the values of $x_i(z,t)$ at the mesh points and two additional unknown parameters. This allows two types of approximations for the intrinsic behavior: triangles and fronts, as e.g. in Fig.1.

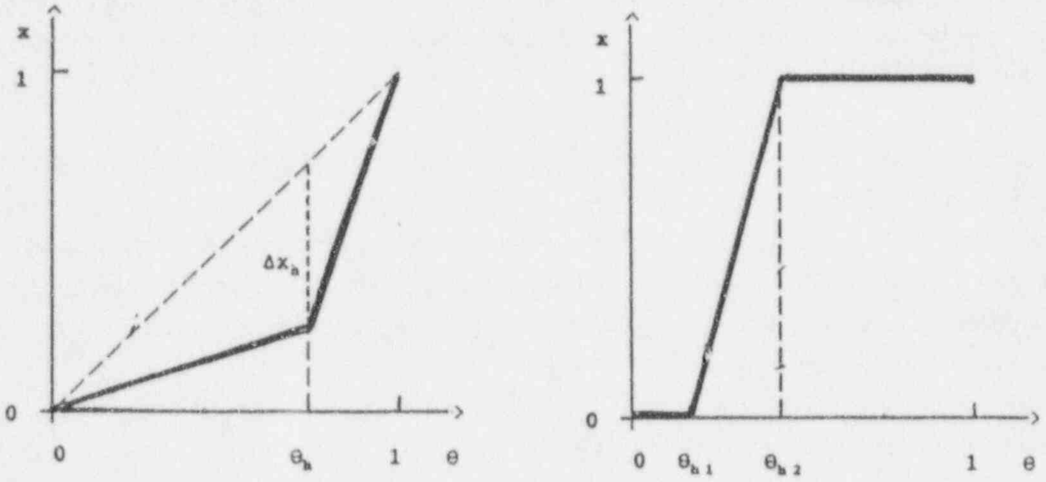


Figure 1. Normalized piecewice linear approximations of PLIM with two unknown parameters

PLIM has useful properties:

- It preserves the shape of the propagating distribution in the sense that when the approximation of the PLIM type is propagated from one mesh cell to another with a constant velocity it can be reproduced exactly and no additional numerical errors occur as the consequence of the propagation[9,10].
- Any distribution at the boundary of the mesh cell can be reasonably accurately approximated.
- Conservation of x in the mesh cell can be satisfied.
- Overshoots and uncontrolled strong variations can be avoided.
- Also the propagation of a front entirely within a mesh cell is described.
- It uses the values of only one mesh interval. Therefore, no extra schemes are needed for the end points or for the discontinuity points of the z-interval. This is a very useful property in the system models, because then the flow networks can be constructed freely but the high accuracy and good applicability to parallel computation can be achieved.

- A good convergence is achieved in solution of any flow conditions of a flow path.
- As a characteristic method, the boundary conditions for a multi-phase flow can be easily defined.

The solver CFDPLIM solves the above system of N flow equations in an arbitrary hydraulic network, which is composed freely of nodes and one-dimensional flow paths. In addition to the normal input, the user have to define the terms of the equations in the own subroutines. The terms are composed of functions of **u** , z , and t. The nodes with finite volumes have to be defined correspondingly. The boundary conditions of the flow paths are defined also with functions.

Because the numerical diffusion has been eliminated, the effects of diffusion and diffusion-like mixing are not included at all in the original version. Only convection has been described. Also this situation obligates to approximate real systems. Therefore, the algorithms of CFDPLIM have been developed to take into account general terms of the second order in respect of the spatial coordinate in order to model diffusion-like mixing.

When one considers hydraulic systems, the above partial differential equations are not sufficient as such to describe hydraulics. The description of hydraulic discontinuity behaviors, such as water levels and shock waves, have to be included. The local linearization of the terms, as well as, many phenomena, such as carry-over, carry-under and surface evaporation, presuppose that the discontinuities are explicitly modeled. Appearing and disappearing of discontinuities in any point of the network requires own description and own particular methods.

The objectives of the development work of CFDPLIM, apart from those satisfied already by the use of PLIM, are

- It solves the unknown quantities when the terms of the equations are correctly defined as functions.
- The geometry of the network is not restricted with any manner. However, the components of the network can be arranged in particular orders to produce better convergence.
- The order N of the equations is arbitrary and it can be different in different parts of network. If N is large and all of the variables are not strongly coupled, the system of the equations can be divided into sub-systems.
- Parallel computation can be utilized in evaluation of the function values. Hence rather complicated forms of the terms of the equations may be used.
- As few algorithms as possible are employed in CFDPLIM. Hence diffusion-like terms and discontinuities are assumed to be a part of the normal computation required for the mesh cells. Also this property offers opportunity to apply parallel computation efficiently.

The properties of CFDPLIM are being heavily tested. Some special features of the discontinuity modeling are still under development work.

Conventional numerical algorithms have difficulties in simulating the transport of sharp fronts in the coolant channels, e.g. of a local boron dilution front. In the boron dilution analyses carried out with HEXTRAN the dilution slugs have been simulated directly to the core inlet in order to minimize the effect of numerical diffusion which tends to reform the boron dilution front into a ramp. However, some numerical diffusion occurs also during the propagation of the diluted slug into the core, especially if the coolant velocity is low. If the time-steps are lengthened numerical dispersion can occur.

CFDPLIM has been implemented into HEXTRAN[11] and the new version of the code is here called HEXTRAN-PLIM. In the following a numerical example, in which boron has no feedback to neutronics, is given where the power of the new method can be seen. A test run for the propagation of a sharp boron front in a vertical channel was carried out with the new HEXTRAN-PLIM code[11]. In reality there occurs some physical diffusion-like mixing in the propagating boron front especially in large open parts of the circuit. These effects can be included in CFDPLIM when best estimate calculations are needed. However, inside the reactor core these mixing effects are small.

The same boron dilution problem was also solved with the widely used commercial hydrodynamics code Phoenics version 2.0. The Phoenics calculations were made with two different time-steps, 0.05 and 0.005 seconds, and with two different spatial discretizations which included 10 or 100 spatial nodes. The HEXTRAN-PLIM calculation was only made with the longer time-steps (0.05 seconds) using 10 spatial nodes.

The initial concentration of boric acid was 7 g/kg (grams boric acid per kg coolant). At time 1.0 seconds the concentration at the inlet of the reactor core was changed to half of its original value i.e. 3.5 g/kg during 0.001 seconds. The concentration was then kept constant during the rest of the calculation. The velocity of the coolant in the reactor was approximately 3.55 m/s. Thus the whole boron front should propagate through the core (length 2.44 m) in 0.69 seconds preserving its original shape. The average concentrations of boron in the lowest and highest nodes were plotted as functions of time.

The solution obtained with the new HEXTRAN-PLIM code is shown in figure 2a. The solution is very accurate and the shape of the propagating front is preserved through the channel. No numerical dispersion or diffusion affect the solution and the minor smoothness of the solution is due to the fact that the average node values and not the calculated mesh point values of the boron concentration are shown. The use of shorter time-steps did not change the results of the HEXTRAN-PLIM code.

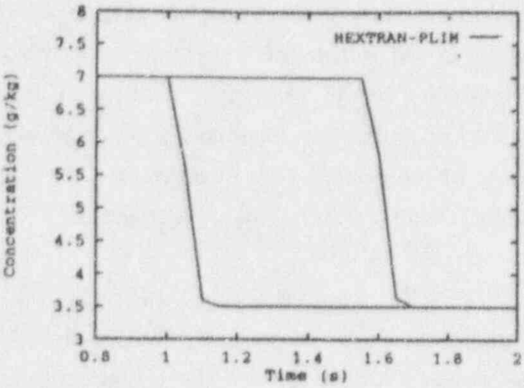


Figure 2a: Boric acid concentration at the inlet and outlet nodes of the reactor core calculated with HEXTRAN-PLIM: Δz = 0.244 m, Δt = 0.05 seconds.

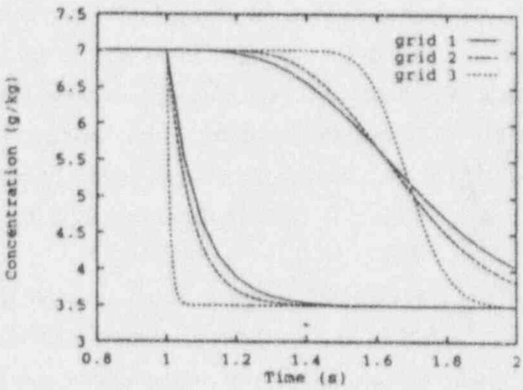


Figure 2b: Boric acid concentration at the inlet and outlet nodes of the reactor core calculated with Phoenics.

Equal grid 1: Δz=0.244 m, Δt=0.05 s

10 times denser grid 2: Δz=0.244 m, Δt=0.005 s

100 times denser grid 3: Δz=0.0244 m, Δt=0.005 s

In figure 2b the results of the Phoenics calculations are shown. It can clearly be seen that numerical diffusion affects the solutions and that the error is reduced by the use of a denser grid. The obtained result was expected and very typical for any conventional numerical hydrodynamics code. The result was significantly improved when the number of mesh cells per time-step was increased to 100, but the effect of numerical diffusion was still very clear. The solution was not as good as the one obtained with HEXTRAN-PLIM, even though a 100 times denser calculation grid was used. It should be noticed that the nodes in the Phoenics calculation with grid 3 have the length 0.0244 m and thus the curve in figure 2b cannot be directly compared to the HEXTRAN-PLIM curve in figure 2a.

It can be concluded that the PLIM algorithm is very accurate for solving problems with propagating fronts. It should be noticed that the numerical errors also affect the transport of the voids in the coolant and the enthalpy of the coolant. These errors are, however, harder to detect since the void fraction and the enthalpy are changed by heat flux from the fuel rods, when the coolant moves through the flow channel.

The entire circuit model using CFDPLIM is under development work.

4. COMPARISON OF HEXTRAN-PLIM AND THE ORIGINAL HEXTRAN IN DIFFERENT FLOW CONDITIONS

A comparison of the HEXTRAN and HEXTRAN-PLIM codes has been carried out in different flow conditions. Two test cases of local boron dilutions in a VVER-440 reactor core were simulated[11]. In the first simulation normal flow conditions were assumed and in the second, more challenging simulation, near natural circulation conditions were assumed in the core.

4.1. Normal Flow Conditions

In the first test calculation a diluted slug of the volume equal to the coolant volume of the reactor core i.e. 6.7 m^3 was set to move through the reactor core. The reactor was initially operating at 50 percent of its nominal thermal power (687.5 MW) and the coolant flow through the reactor core was close to nominal (7300 kg/s). The concentration of boric acid in the whole reactor was initially 7.6 g/kg, a typical value for the beginning of a fuel cycle situation. Only a reasonable mild dilution was used in the test cases in order to avoid boiling of the coolant. The calculation was made with 20 axial nodes and with only a few channels in the core. The time-step used in the calculations was 0.02 seconds.

At 1.0 seconds a slug of diluted water entered the reactor core and the boric acid concentration at the inlet decreased linearly during 0.02 seconds to a value of 7.3 g/kg. At 1.7 seconds, when the edge of the diluted slug had reached the outlet of the reactor core, the boric acid concentration at the inlet of the reactor increased linearly to its original value during 0.02 seconds.

The calculated boric acid density at the inlet node of the reactor core as a function of time is shown in figure 3a. The results of HEXTRAN and HEXTRAN-PLIM are in a very good agreement with each other. Numerical diffusion does not significantly affect the results although the "corners" in the solution obtained with the HEXTRAN code are slightly rounded. The coolant density in the first calculation node remains almost constant during the transient since the increase of enthalpy in the first calculation node is insignificant compared to the inlet enthalpy of the reactor. In figure 3b the calculated boric acid density at the outlet node of the reactor core is shown. There the boric acid density has significantly decreased due to the decreased density of water. The calculated results of both codes are qualitatively the same. However, both numerical diffusion and dispersion affect the solution obtained with HEXTRAN, while the HEXTRAN-PLIM solution is very stable. The original shape of the disturbance is modified by the increased power to coolant, but otherwise the shape is very well preserved. The oscillations of the boric acid density produced by the HEXTRAN code are clearly nonphysical but qualitatively the result is very close to the true solution.

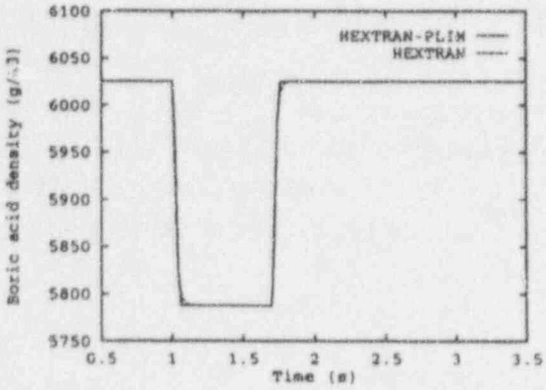


Figure 3a: Boric acid density at the inlet node of the reactor core during a boron dilution.

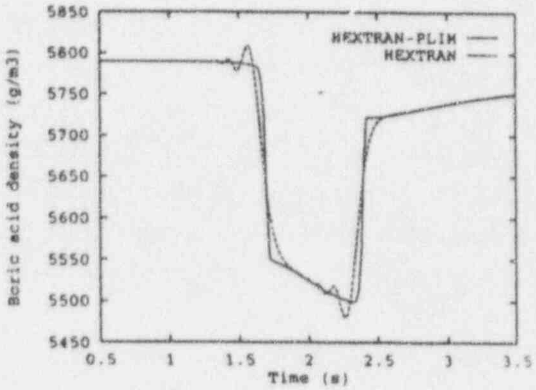


Figure 3b: Boric acid density at the outlet node of the reactor core during a boron dilution.

Generally, when purely linear numerical methods are used, numerical dispersion cannot be completely removed without an increase of numerical diffusion. The numerical algorithm used here in HEXTRAN is specially designed to minimize the total numerical error and thus both forms of numerical errors occur in the solution.

The produced fission power peaks calculated with HEXTRAN and HEXTRAN-PLIM are shown in figure 4. Due to the decreased amount of boric acid in the reactor, the fission power increases and reaches a maximum value (1683 MW) at 1.68 seconds. The results of the two codes are almost identical. The local disturbances of the boric acid densities in the HEXTRAN solution do not affect the total behavior of the core.

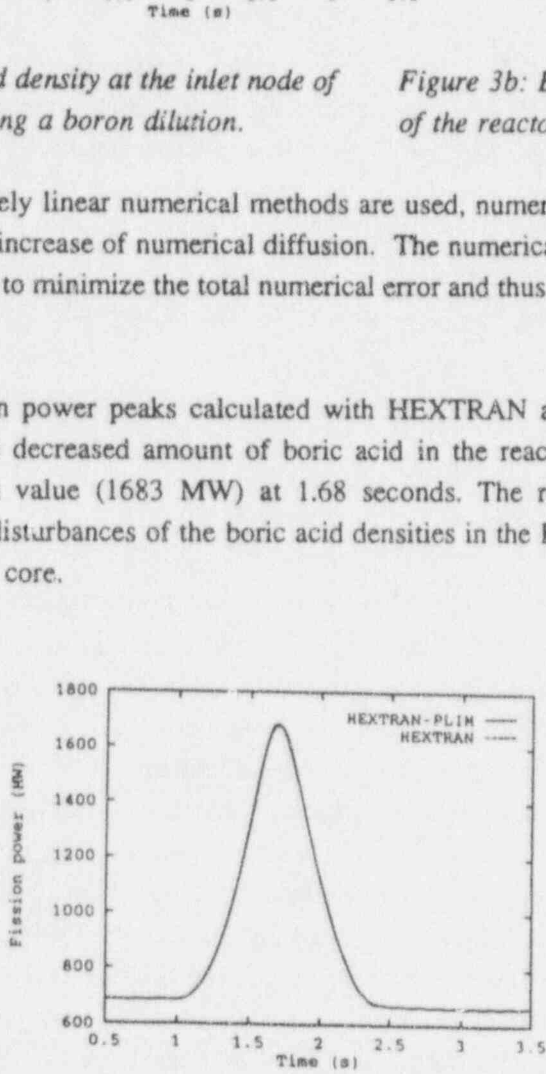


Figure 4: Total fission power during a boron dilution in normal flow conditions.

The calculated results of the first boron dilution test case can be considered to be in a very good agreement with each other. The HEXTRAN-PLIM code gives very accurately the same results as the HEXTRAN code, although the numerical solution of the hydrodynamic flow produced by HEXTRAN-PLIM is more accurate and close to the ideal solution. This confirms the good performance of the HEXTRAN code under normal flow conditions.

4.2. Natural Circulation Conditions

In the second test case of a boron dilution accident simulation, near natural circulation conditions were assumed. Thus the second case is numerically much more challenging than the first case. There are many conflicting requirements to the spatial and time discretizations and, as a consequence, the hydraulics calculational grid applied in the calculation in this test case is very far from fulfilling the Courant criterium.

The initial reactor power was 1 percent (13.75 MW) of the nominal thermal power and the mass flow through the reactor was initially 742 kg/s, which is approximately 10 percent of the nominal. The volume of the diluted slug was decreased to 50 percent of the core coolant volume. The same initial and diluted boric acid concentrations as in case 1 (7.6 and 7.3 g/kg coolant) were used. The total time of the transient simulation was 20 seconds and at 1.0 seconds the diluted slug entered the reactor core. At 4.5 seconds the boric acid concentration at the inlet of the reactor was increased to its original value.

The calculated boric acid densities at the inlet node of the reactor core are shown in figure 4a. The results of HEXTRAN show clear signs of numerical diffusion, while the solution obtained with HEXTRAN-PLIM is very accurate. The decrease of the boric acid density is initially faster in the solution obtained with HEXTRAN because linear approximation between the mesh points is used to calculate the node value. It can be noted that both solutions conserve very well the total mass of the boric acid which can be approximatively seen from figure 5a using the fact that the area below the curves should be the same.

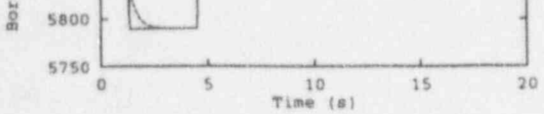


Figure 5a: Boric acid density at the inlet node of the reactor core during a boron dilution.

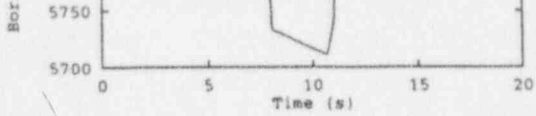


Figure 5b: Boric acid density at the outlet node of the reactor core during a boron dilution.

In figure 5b the calculated boric acid densities at the outlet node of the reactor core are shown. Here the HEXTRAN solution is markedly smoothened due to numerical diffusion. The HEXTRAN-PLIM solution is again nearly exact. The boric acid mass is conserved in both solutions. The most serious consequence of the smoothening of the HEXTRAN results is that the reactivity worth of the slug is decreased. Due to the finite volume of the slug, the smoothenings of the edges of the slug are combined and the maximum dilution level is not achieved at all in the upper part of the core. As expected the fission power peak produced by HEXTRAN remains clearly smaller than the peak produced by HEXTRAN-PLIM. Also the energy release is essentially smaller as can be seen in figure 6.

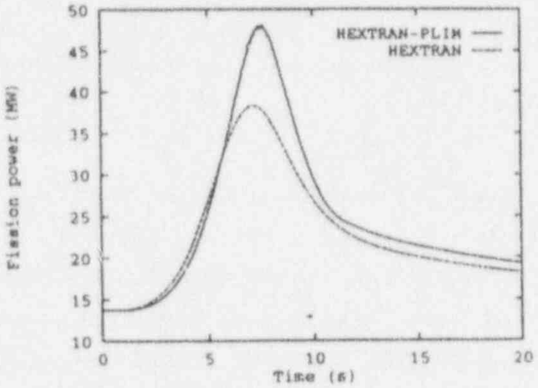


Figure 6: Total fission power during a boron dilution in natural circulation conditions.

In earlier analyses made with HEXTRAN in natural circulation conditions, this prominent effect on the numerical errors in simulation of finite slugs has been taken into account by using conservative assumptions. The volume of the slug has been enlarged so that the minimum concentration of boric acid has been achieved even in the last nodes at the outlet of the reactor. Using this same conservative assumption here, the power peak was 230 percent greater than the one calculated with HEXTRAN-PLIM. Increasing the volume of the diluted slug with 30 percent was enough to create a power peak in the HEXTRAN calculation as large as in the HEXTRAN-PLIM calculation with the original slug volume. Then the minimum concentration was achieved in the central part of the core but still not in the last nodes at the outlet of the reactor core. Thus the earlier mode of action in the HEXTRAN analyses has guaranteed the conservativity of the results but there has been considerable overconservatism in the analyses.

5. CONCLUSIONS

With the three-dimensional HEXTRAN code, reactor dynamics accident analyses such as RIA, ATWS and boron dilution analyses can be carried out.

The reactor power and the propagation of a diluted boron slug are coupled with each other and they have to be solved simultaneously in the reactor core. The rigid discretization mesh and the time-steps dictated by neutronics have to be used, since neutronics is very sensitive to any change. In natural circulation conditions, it is not possible to obtain the velocities of the boron fronts close to satisfying the Courant criterium and significant numerical diffusion appears when normal numerical methods are employed in the core hydraulics. Then the boron fronts are erroneously smoothened and the reactor power remains too small. We showed an example, where the diluted boron slug fills half of the core volume and the numerical diffusion error is tried to compensate with enlarging the slug volume. Then the diluted volume has to be enlarged with 30 percent in order to produce the correct power peak.

The implementation of CFDPLIM to HEXTRAN eliminates the numerical diffusion and dispersion from the thermal hydraulics solution. This application guarantees that conservative accident analyses of local boron dilutions can be done even in numerically difficult flow conditions without additional conservative factors due to the numerical method. The comparison calculations also show that the conservative assumptions used in the some earlier analyses may lead to a 230 percent over-conservative result in respect of the power peak. Besides the core calculations CFDPLIM has potentiality into the circuit calculations, viz. in analyzing complex accidents, the accurate numerical methods offer the most effortless and most reliable way to estimate consequences of the accidents.

ACKNOWLEDGEMENTS

The authors would like to thank T. Stenius for his tasks in the implementation work of CFDPLIM in the HEXTRAN code and for carrying out the test calculations with the new code version.

The work has been jointly funded by VTT and Finnish Centre for Radiation and Nuclear Safety (STUK).

REFERENCES

1. Siltanen, P., Antila, M., Kyrki-Rajamäki, R. & Vanttola, T. Analysis of core response to the injection of diluted slugs for the Loviisa VVER-440 reactors. Proceedings of the OECD/NEA/CSNI Specialist Meeting on Boron Dilution Reactivity Transients. State College, PA 18 - 20 October 1995.

2. Kyrki-Rajamäki, R. Three-dimensional reactor dynamics code for VVER type nuclear reactors. Espoo: Technical Research Centre of Finland, 1995. 51 p. + app. 79 p. (VTT Publications 246).

3. Kaloinen, E., Siltanen, P. & Teräsvirta, R. Two-group nodal calculations in hexagonal fuel assembly geometry. Proceedings of a Specialists' Meeting on The Calculation of 3-dimensional

Rating Distributions in Operating Reactors. Paris 26 - 28 November 1979. Paris: OECD/NEA. 1980. Pp. 111 - 128.

4. Rajamäki, M. TRAB, a transient analysis program for BWR, Part 1. Principles. Helsinki: Technical Research Centre of Finland, Nuclear Engineering Laboratory, Report 45, 1980. 101 p. + app. 9 p.

5. Miettinen, J. Development and assessment of the SBLOCA code SMABRE. Specialists' Meeting on Small Break LOCA Analyses in LWRs. Pisa 23 - 27 June 1985. Pisa: Dipartimento di Costruzioni Meccaniche e Nucleari Università di Pisa, 1985. Vol. 2, Pp. 481 - 495.

6. Kyrki-Rajamäki, R. Validation of the reactor dynamics code HEXTRAN. Helsinki: Finnish Centre for Radiation and Nuclear Safety, 1994. 24 p. (STUK-YTO-TR 69).

7. Rajamäki, M. & Saarinen, M. PLIM shape-preserving characteristics method for flow equations. International Topical Meeting on Advances in Mathematics, Computations and Reactor Physics. Pittsburgh, PA 28 April - 2 May 1991. La Grange Park: American Nuclear Society, 1991. Pp. 12.1 1-1 - 12.1 1-13.

8. Rajamäki, M. & Saarinen, M. Accurate one-dimensional computation of frontal phenomena by PLIM. Journal of Computational Physics, 1994. Vol. 111, nro 1, pp. 62 - 73.

9. Rajamäki, M. & Saarinen, M. A numerical study of the withdrawal and return of water following a volcanic eruption at sea. Communications in Numerical Methods in Engineering, 1994. Vol. 10, nro 6, pp. 461 - 468.

10. Saarinen, M. Dynamic model for horizontal two-phase flow predicting low head flooding. Numerical Heat Transfer, Part A, Applications, 1994. Vol. 26, nro 4, pp. 471 - 482.

11. Stenius, T. Implementing the characteristics-based numerical algorithm PLIM to the flow model of the reactor dynamics code HEXTRAN. Espoo: Helsinki University of Technology, 1994. 80 p. (Master's thesis).

A BORON TRANSPORT SIMULATION USING RETRAN-03 CONTROL SYSTEM MODELS FOR ANTICIPATED TRANSIENT WITHOUT SCRAM (ATWS) EVENT

by

J.S. Miller *
SCIENTECH, Inc.
11140 Rockville Pike, Suite 500
Rockville, MD. 20852

and

Ching N. Guey *
Florida Power & Light

SPECIALIST MEETING ON BORON DILUTION REACTIVITY TRANSIENTS

State College, PA. ; USA
18 th - 20 th October 1995

Hosted by

U.S. Nuclear Regulatory Commission
Penn State University, Nuclear Engr. Dept.
TRAC Users Group

* Original work was performed for Gulf States Utilities at River Bend Station

ABSTRACT

This paper is being presented at the Specialist Meeting on Boron Dilution Reactivity Transient. The meeting is being held on October 18, 19 and 20 in State College, PA.

This paper presents the results of Anticipated Transient Without Scram (ATWS) analyses performed for River Bend Station (RBS) in 1987. RBS is a Boiling Water Reactor/6 (BWR/6) with 624 fuel assemblies and an inside reactor vessel diameter of approximately 218 inches. During postulated ATWS events, important operator actions and system design considerations, among other things, include: (1) stand pipe lower plenum injection effectiveness, (2) water level control, (3) enrichment of Boron-10, (4) ability to maintain adequate suppression pool temperature within the limits based on heat capacity considerations, and (5) ability to maintain and achieve the cold shutdown conditions. This work considers the first four items for RBS ATWS analysis by incorporating a new boron transport model in the RBS RETRAN model that represents a more realistic analysis based on local mass balance of the boron between each volume and its adjacent neighboring volumes.

The boron transport model assumes that the boron flow at each junction is proportional to the junction liquid flow and the associated boron concentration from which the junction emanates. Control system models in the RETRAN program were used as building blocks for the boron transport model. RETRAN03-PRE34 version with a hardwired modification performed by Energy Incorporated (EI) to treat boron absorption was used for this work. This approach builds upon the work by C. G. Metloch et al, "A Methodology for the Calculation of Boron Concentration in Boiling Water Reactors." presented at the Third International RETRAN Meeting[5]. This study extends Metloch's work to a complete plant ATWS analysis including water level reduction, one-dimensional kinetics and thermal-hydraulic feedback. Many of the complications introduced by the ATWS computer simulation make this study challenging and unique. In addition, the following features were included in the new boron transport model reported in this paper:

1) Implicit Solution Scheme
2) Boron Mass Balance Tracking
3) Flow Reversal Situation

The overall results based on the new boron transport model indicate that with 65% enrichment of B-10, and water level controlled at TAF+5ft and TAF+10ft, the suppression pool temperature can be maintained within heat capacity temperature limit of the suppression pool.

The use of 65% B-10 enrichment provided significant cost savings compared with using 90% enrichment. Initial ATWS analyses indicated that a 90% enrichment would be required to attain the specified goals. With more refinement in the boron tracking models, a significant reduction in cost was achieved. In addition, the use of 65% B-10 enrichment eliminates the need of heat tracing on the SLCS. The RBS ATWS analysis also provided confidence that maintaining water level at TAF+5ft is sufficient to shut down the reactor avoids unnecessary actions to use Automatic Depressurization System (ADS).

ACKNOWLEDGMENTS

The authors would like to thank Energy Incorporated for providing significant input on the initial work on model development and analysis.

Also, appreciation to Kim Jones for typing this manuscript and to Gulf States Utilities, especially Mel Sankovich, Jim Booker and Jim Deddens for providing their support. Gratitude is also extended to SCIENTECH, Inc. whom provided the support to present this paper.

CONTENTS

ILLUSTRATIONS

Figures

Tables

Section 1
PURPOSE

Emergency Procedure Guidelines (EPGs)[l] for Anticipated Transient Without Scram (ATWS) events have undergone close scrutiny within the nuclear industry to better define the expected sequence of events and recommended operator actions This work was performed at Gulf States in 1987, in part, to support plant-specific Emergency Operating Procedures (EOPs) for ATWS scenarios The primary objectives of this work are (l) to determine if the boron delivery point which is through a stand pipe in the lower plenum (i.e., the boron is in solution as sodium pentaborate) is appropriately located to shut down the reactor in the event of a worst case ATWS, (2) to evaluate the effects of different enrichments of Boron-10 on an ATWS event and decide which enrichment level meets the basic requirements of this work, (3) to determine the merits of maintaining water levels higher than the top of active fuel (TAF) presently recommended by the BWR Owner's Group EPGs, and (4) to demonstrate the ability of standby liquid control system (SLCS) to achieve hot shutdown before suppression pool temperature exceeds Technical Specifications limits. As part of this study a new boron tracking model was developed to track the mass flow of the boron throughout the reactor vessel and keep track of the boron mass.

Because of practical constraints, this work focused only on determining whether the suppression pool temperature would stop increasing when the reactor power reaches 5% or lower. The 5% of rated power is the rated heat removal capacity of residual heat removal (RHR) system in suppression pool cooling mode.

This paper summarizes the assumptions and models used and discusses the results based on various boron tracking models.

Section 2
BACKGROUND

An Anticipated Transient Without Scram (ATWS) is an expected operational transient (such as a main steam isolation valve (MSIV) closure, a loss of feedwater, loss of condenser, or loss of offsite power) which is accompanied by a failure of the reactor trip system to shut down the reactor.

As part of the final ATWS rule, a 86 gpm Boron Injection Capability or Equivalent is required. As a result of these analyses enriched Boron-10 at 65% enrichment in nominal 9% sodium pentaborate solution is used at RBS to satisfy the 86 gpm equivalency. This design avoids the difficulty of rerouting the SLCS piping to the high pressure core spray (HPCS) lines. The enriched Boron-10 is injected into the reactor vessel lower plenum by manually initiating SLCS.

The analyzed ATWS event simulated for these evaluations is a MSIV closure from 100% power. The major tasks involved in the best-estimate analysis of several ATWS scenarios included modeling the numerous operator responses, developing the large and complex model consistent with the computer capacity, representing boron mixing in the pressure vessel, and simulating the mixing of subcooled water injected into a two-phase region.

Since these transients are numerically difficult, the prerelease version of RETRAN-03 was used to take advantage of its significantly faster computational speed than RETRAN-02.

It is noted that the work spanned more than a two-year period due to the complex and difficult decisions involved in selecting operator actions, choosing various assumptions and developing best-estimate boron tracking models.

Because the effectiveness of SLCS and the variation of water level control represent two most important aspects of mitigating ATWS scenarios, the following sections provide a basic description of the SLCS and the water level control for ATWS events.

GENERAL DESCRIPTION OF THE SLCS

The SLCS is designed to inject into the reactor a sufficient quantity of a neutron-absorbing solution (sodium pentaborate) so as to achieve reactor shutdown, independent of control rod action, under the most reactive core conditions. Some of the design parameters of the SLCS with respect to determining the amount of time required to shut down the reactor include: (1) SLCS pumping rate, (2) the neutron absorber's mixing efficiency once it is injected into the reactor vessel, and (3) weight percentage of the absorber in the SLCS solution.

The SLCS pumping rate depends on the capacity, the number of SLCS pumps used, and the piping alignment. The mixing efficiency of the boron depends on the core flow. The weight percentage of the neutron absorber (i.e. B-10) depends on the concentration of sodium pentaborate in the solution, the enrichment of the B-10, and the mass of water in the reactor pressure vessel.

The current SLCS design and specified sodium pentaborate boron enrichment was revised to comply with the ATWS rule, provide additional margin for the removal of heat tracing requirement from the RBS Technical Specifications, and provide RBS Emergency Operating Procedures (EOPs) margin for not exceeding the heat capacity temperature limit (HCTL) of the suppression pool during a worst case ATWS event.

GENERAL DISCUSSION OF WATER LEVEL REDUCTION

The core power in a BWR can be controlled by varying core flow. For example, for the worst case ATWS event, the recirculation pump trip would immediately reduce the reactor core flow and hence reduce the reactor power. Following the trip of the recirculation pumps, the reactor is in natural circulation mode. The driving head to establish natural circulation is created by the fluid density difference inside versus outside the shroud, and inside versus outside the active channels Figure 1 shows typical BWR natural circulation paths.

The two-phase fluid inside the active channel has a lower density than the water in the bypass region and outside the shroud. The natural circulation driving density head for the path including downcomer, Jet pumps and core inlet, is determined by the reactor water level, reactor power and core void fraction. Natural circulation driving head and flow in this path drop as reactor water level drops. The reduced core flow produces more voids in the core and thus further suppresses power generation. This power suppression is achieved, if needed, by manually lowering reactor water level following an ATWS.

EPGs Revision 4[1] contains a CONTINGENCY #5 for Level/Power Control In CONTINGENCY #5, lowering water level in the reactor pressure vessel to TAF is recommended if reactor power is above Average Power Range Monitoring (APRM) downscale trip setpoint (5% for River Bend) or cannot be determined. It is one of the objectives of this study to determine whether lowering water levels to higher than TAF would be sufficient to comply with heat capacity temperature limit as specified in EPGs Revision 4 and maintain adequate mixing in the lower plenum to allow boron to be circulated into the active core.

Section 3
EVALUATION

The ATWS event analyzed for this paper is a main steam isolation valve (MSIV) closure from 100% power with failure to scram. This event was determined in NEDO-24222 to be the worst case ATWS. It is recognized that there are many other possible ATWS scenarios, but the worst case ATWS was used to test the various methods of controlling the ATWS event. A constraining factor in such an event is the suppression pool (SP) Heat Capacity Temperature Limit (HCTL) as prescribed by EPGs Revision 4. The SP temperature limit prescribes the temperature above which a 100% steam condensation in the SP cannot be assured during a complete blowdown of the Reactor Pressure Vessel (RPV).

Following an MSIV closure, the vessel water level decreases due to collapsing of voids. At water level 3, vessel water level setpoint setdown is initiated. When the reactor vessel pressure reaches approximately 1100 psig, recirculation pump trip occurs. Because of continuing high power, pressure of the RPV is increasing. The Safety Relief Valves (SRVs) open to allow steam into the suppression pool. The suppression pool is gradually heated up by the steam released from SRVs. The residual heat removal (RHR) system is then actuated with some time delay. In addition, the SLCS and water level control are manually initiated. The type of scenario described above was analyzed in this study by varying the following assumptions and models:

- o Water level to be Controlled

  Three water levels at which the operator would maintain the reactor were analyzed Top of Active Fuel (TAF), TAF+5ft, and TAF+10ft.

- o Operator Actions

  The time delays associated with the various operator actions were varied. These operator actions were injection of SLCS, reduction of water level, and initiation of RHR system suppression pool cooling mode.

- o Thermal-Hydraulic Modeling

  Slight nodalization and modeling associated with injection of subcooled liquid were made to alleviate oscillation problems.

- o Boron Tracking Model

  Four boron tracking models with different levels of detail and sophistication were included in the ATWS analyses using RETRAN.

REACTOR PHYSICS MODEL

The RETRAN-03 analysis of a turbine trip ATWS for the GSU River Bend plant invoked the space-time kinetics model to calculate the power responses. In the absence of a reactor scram, ultimate reactivity control is assumed to depend on Boron-10 injection into the core vessel. Thus, the cross-sections used in the analysis must also include the effects due to the presence of boron. Including such an explicit boron dependence in the cross-sections required a code modification since the standard cross-section model includes only coolant density and fuel temperature dependencies. For the River Bend ATWS analysis, the basic cross-section set was developed with no boron present and a correction to the absorption cross-section applied to account for boron when present. More information on this development is provided in Reference 3.

OPERATOR ACTIONS

When the suppression pool temperature reaches 95°F, operators were required to initiate RHR suppression pool cooling mode. When the suppression pool temperature reaches 110°F as a result of ATWS, the following actions should be considered based on EPGs:

- o lower water level, and
- o initiate SLCS

Since these actions are all manually initiated and all involve critical decisions on the part of operators, some delay would be associated with each of the three actions. Table 1 presents two operator action models considered in this analysis. Operator Action 1 was based on preliminary information. Operator Action 2 was based on a survey of the shift supervisors at RBS which provided best estimate response times for the operator actions.

THERMAL-HYDRAULIC MODELING

The nodalization of the RPV and related injection systems is essentially the same as that used for Reference 3. The only modifications made included the following:

- o Subcooled Injection Model

  The original subcooled injection model used in Reference 3 was developed to circumvent expected oscillation problems when subcooled water is injected into a region filled with a two-phase mixture. This model was used for cases 1, 2 and 3 of the ATWS analysis but was later replaced by artificially lowering the feedwater (FW) injection point to lower elevations.

- o Constant Pressure Safety Valve Model

  Previous experience with RETRAN indicated that at low vessel water levels, unrealistic power spikes and other unstable numerical behavior resulted when the SRVs were allowed to cycle open and closed. To alleviate this problem, "constant-pressure" SRVs were modeled. The constant-pressure SRV model consists of holding the steam dome pressure constant by modulating the SRV with the highest pressure setpoint while holding open the other SRV banks with lower pressure setpoints. As the system power decreased, the pressure continuously decreased below the highest pressure setpoint even with the corresponding SRV bank completely closed. When the steam dome pressure fell to the next pressure setpoint, the pressure was maintained at this point by modulating the SRV bank while holding open the remaining SRV banks with lower pressure setpoints. This process was repeated until the lowest pressure setpoint was reached and the pressure was maintained at this point by modulating the last SRV bank. Sensitivity studies were performed to provide evidence that this simplification produced valid results.

- o Nodalization

  The plant model consists of a nodalization almost identical to that used in Reference 3. The only difference is the division of the lower plenum into two volumes to more realistically model the thermal-hydraulic phenomena (e.g., split fraction between bypass flow and active core flow). Figure 2 shows the RETRAN nodalization diagram for the River Bend ATWS model.

BORON TRACKING MODELS

In this section four boron tracking models used for the River Bend ATWS analyses are discussed These methods are presented in ascending order of complexity. The four boron tracking models are 1) Empirical Correlation based on mixing coefficient, 2) Mechanistic Model using time dependent DELAY and LAG control ,blocks, 3) Ad-Hoc Fix to Model 2, and 4) Mechanistic Model based on local mass balance. A summary of these models is presented in Table 2.

Empirical Correlation Based on Mixing Coefficient

An empirical correlation obtained from mixing tests performed on a 1/6-scale BWR model was used to calculate the boron concentration in the core. The mixing test results provided a mixing coefficient n, defined as the ratio between solution concentration in the core and that over the entire vessel assuming perfect mixing.

The mixing test correlation is applicable for values of f (where f is fraction of rated core flow) from 0.05 to 0.20. For f between 0.20 and 0.35 the value of n obtained for f equal to 0.20 was used as suggested in Reference 5. For values of f less than 05, Reference 6 showed that inadequate mixing in the lower plenum would not allow boron into the core.

Since actual core flow rates vary during SLCS injection, the correlation was reformulated. This was done by using the control system blocks to perform the integration of n for each time step.

The total core boron mass was obtained by multiplying the core liquid inventory and the core boron concentration. The boron mass in each of the 12 active core regions is obtained by evenly distributing the total core boron mass.

Mechanistic Model Using Time-Dependent DELAY and LAG Control Blocks

The methodology is based to a large extent on that proposed by Metloch ,et. al., [5]. This methodology utilizes control blocks to calculate the boron concentration in fluid volumes. The boron reactivity worth is calculated from the core boron concentration and is used with the RETRAN 1-D kinetics to provide shutdown reactivity. The following represent major deviations of the River Bend ATWS analysis from that reported in Reference 5:

- o a full reactor system simulation of the ATWS scenario was analyzed
- o 1-D kinetics was used
- o 12 core regions were used
- o transport of boron mass between core volumes was modeled using time dependent DELAY control blocks

Figure 3 shows a simplified boron concentration model adapted from Reference 5 for the ATWS analysis of the River Bend Station.

Ad-hoc Fix to the Mechanistic Model

The model described in the previous subsection introduced significant oscillations of boron concentration and power. To correct the inadequacy caused by the numerical oscillations, an ad-hoc fix to the model was implemented in the revised model. The ad-hoc fix consists of specifying a non-negative values as a minimum (10E-6) in the control blocks that calculate the boron mass increment at each time step. In addition, the water injection was modified to provide a less drastic change in water addition to the vessel.

A Mechanistic Model Based on Local Mass Balance

The ad-hoc fix as described in previous subsection yielded slightly biased results because of the artificial minimum values imposed on control blocks calculating the boron mass increment. The artificial imposition of minimum boron mass increment at each time step biased the oscillation in such a way that overall boron mass undergoes a step increase during power oscillations. Therefore, a boron model based on local mass balance is developed to accurately track the boron transport. The following assumptions provide the basis for the new mechanistic model.

- o Instantaneous mixing of boron takes place in a volume. The boron influx from the upstream volume is mixed uniformly within a time step. This may be slightly conservative because it actually takes some time for boron to achieve uniform mixing. Boron concentration would be higher closer to the source, i.e., core volumes would see higher concentration than that assumed for uniform mixing.

- o Boron flow is equal to the product of concentration and junction liquid flow. The flow of boron is treated as a slug flow, being carried with the liquid.

Based on the above considerations, the following expressions are derived for boron transport modeling:

B(i,t-delta t) - Boron Mass in Volume i, at time t-delta t
B(i,t) - Boron Mass in Volume i, at time t
B(i-l,t) - Boron Mass in Volume i-l, i.e., upstream volume, at time t
J(i+l,t) - Junction Liquid Flow for Junction i+1, at time t
J(i,t) - Junction Liquid Flow for Junction i, at time t

M(i,t) - Liquid Mass for Volume i, at time t
M(i-l,t) - Liquid Mass for Volume i-l, at time t
delta t - Incremental Time, i.e., time step

Junction i is inlet to Volume i, Junction i+l is outlet to Volume i+l,

$$B(i,t) - B(i,t-to) + t \times (B(i-l,t)/M(i-1,t)) \times J(i,t) - (B(i,t)/M(i,t)) \times J(i+l,t)$$

or to solve implicitly,

$$B(i,t) = B(i,t-to)+ t \times(B(i-l,t)/M(i-1,t)) \times J(i,t) / 1 + (J(i+l,t)/M(i,t)) \times t$$

For flow reversal situations, other expressions are used.

In addition to the revised boron transport model, control blocks were added to calculate the total boron injected, the total boron in the vessel, and the difference between the two. This provided a boron mass error calculation.

The major features of the revised boron model are:

- o Better tracking of boron transport
- o More reasonable results.

Section 4

ANALYSES PERFORMED AND RESULTS

A total of 12 cases was studied in the River Bend ATWS analysis work. Each case consists of a different combination of water level being maintained, boron enrichment, operator actions, and various boron tracking models. The calculated maximum suppression pool temperatures for various cases studied are presented in Table 3. Table 4 presents a typical timing of major events for the various cases analyzed. It is noted that the sequence of events (mainly, the timing) is identical for all the 12 cases analyzed until the suppression pool temperature reaches 110°F. From then on, the following assumptions would impact the transient response significantly.

- o Operator actions

  Operator actions to initiate the SLCS, to reduce vessel water level, and to initiate RHR suppression pool cooling could have significant impact on the maximum suppression pool temperature.

- o Water Level Being Maintained

  The water level was maintained at TAF, TAF+5ft, and TAF+10ft for various cases studied. Higher water level tended to provide a better mixing but a slightly higher power due to less void. However, water level at TAF did not provide sufficient core flow (i. e., less than 5% of the rated flow) to carry the boron into the core.

- o Enrichment of B-10 and Concentration of Sodium Pentaborate Solution

  Various combinations of B-10 enrichment and solution concentration change the number of B-10 atoms per unit volume. The higher the enrichment of B-10 is, the lower the maximum suppression pool temperature.

A typical analyzed worst case ATWS scenario begins with the closure of the MSIVs and the failure of the control rods to scram. The MSIVs close in approximately four seconds, during which steam flow decreases and pressure begins to rise.

Shortly after the MSIVs shut, pressure will increase enough to lift the SRVs. The steam discharge from the SRVs is directed to the suppression pool (SP) and heats up the pool Thirty (30) seconds (Operator Action Model 1) or 90 seconds (Operator Action Model 2) after the SP temperature reaches 110°F, feedwater (FW) flow is, temporarily terminated to cause the downcomer water level to drop. When the water level falls to the desired level (TAF, TAF+5ft, or TAF+10ft, depending on the case analyzed), FW flow is reinstated as necessary to maintain the level. Two minutes (for Operator Action Model 1) or 72 seconds (for Operator Action Model 2) after the SP temperature reaches 110°F, the SLCS is activated, injecting boron into the lower plenum. The transport of boron through the vessel varies as a different boron tracking model is used for different cases. The RHR system is activated ten minutes (for Operator Action Model 1) or 90 seconds (for Operator Action Model 2) after the SP temperature reaches 95 F. The RHR system provides cooling at approximately 5% of the rated reactor power.

For the cases in which the calculated SP temperature exceeds the heat capacity temperature limit (HCTL) as specified in the GSU EOPs, automatic depressurization system (ADS) operation will be initiated. However, this work did not model the subsequent transient response following an ADS for such cases.

For the cases in which the calculated maximum SP temperatures comply with the HCTL, the analysis was terminated shortly after the SP temperatures started to decrease or when the reactor power dropped to below 5%. All cases are summarized in Table 3.

Cases 1, 11 and 12 provides useful results for comparison. Case 1 represent lowering the water level to TAF, Case 11 represented lowering the water level to TAF + 5 feet, and Case 12 represented lowering the water level to TAF + 10 feet.

Figure 4 presents the comparison of these three cases for core flow percentage versus water level. As shown in Figure 4, the core flow is basically a reflection of power level, i.e., the higher the water level, the higher the power level which corresponds to a high core flow rate. The important point presented in Figure 4 is for the TAF case at approximately 470 seconds into the transients, the core flow drops below 5 percent. Based on Boron mixing test results provided in Reference 6, the core flow below 5 percent does not provide enough lower plenum mixing to allow the sodium pentaborate (being injected through the standpipe into the lower plenum) to be swept into the core. Therefore, with the core flow below 5 percent for standpipe injection plants, the effectiveness of the SLCS is negated due to the lower plenum stratification.

Figure 5 presents the power level with respect to water level and Figure 6 presents the suppression pool temperature with respect to water level. As one can see, the TAF suppression pool temperature will eventually exceed the heat capacity temperature line (HCTL) for the suppression pool due to the stratification of the boron in the lower plenum. It is noted that significant power oscillations were experienced in the cases analyzed when water injection and SRV cycling occurs. These oscillations were not indicated in Figures 4 and 5 for simplifying the presentation.

Section 5
CONCLUSIONS

This work was performed to achieve the following:

- o demonstrate the effectiveness of the SLCS design for RBS
- o perform a sensitivity study of boron tracking models,
- o evaluate the effects of B-10 enrichment,
- o determine the merits of reducing water level to higher than TAF, and

Based on the results of 12 cases presented in Table 3 with various combinations of B-10 enrichments, water level, boron tracking models and operator action models, the following conclusions can be reached

I. Technical Findings

- o The higher the B-10 enrichment is, other things being equal, the more effective the SLCS is in reducing the power, hence the lower the SP temperature.

- o Reducing water level to TAF+5ft and TAF+10ft yields a best-estimate maximum suppression pool temperature of 150 F and 155 F, respectively, for the proposed SLCS design at RBS (i.e., 65% B-10 enrichment, at 9% nominal solution concentration). The case in which water level is maintained at TAF does not have sufficient boron mixing and hence the calculated suppression pool temperature is higher than the case where water level is maintained at TAF+5ft or TAF+10ft.

- o Of the four boron tracking models studied, Model 4 provides a best-estimate simulation of the boron transport in the vessel. Model 1, which is based on the mixing coefficient with a 118-second transport delay, and model 2, which uses time-dependent control system blocks DELAY and LAG, both give very conservative estimates of the SP temperature.

- o The current SLCS design at the RBS, with water level being controlled at TAF+5ft or TAF+10ft, is adequate in limiting the suppression pool temperature within the heat capacity temperature limit specified in EOPs.

- o The RBS ATWS analysis results provide supplementary information to the EPGs Revision 4 for scenarios in which SLCS is successfully injected. It is demonstrated with the current SLCS design either maintaining water level at TAF+5ft or TAF+10ft would provide a sufficient power reduction mechanism to mitigate the SP heatup.

- o By lowering the water level to TAF may for lower plenum SLCS injection plants, negate the effectiveness of injecting the sodium pentaborate into the lower plenum for reducing the power level.

II. Practical Benefits to GSU

1. The RBS ATWS analysis provided significant cost savings in the implementation of ATWS modifications. Instead of using 90% B-10 enrichment, 65% enrichment has been demonstrated to be adequate for satisfying the requirements of the ATWS modifications.

2. The RBS ATWS analysis allowed elimination of heat tracing requirements for the SLCS because of the lower B-10 enrichment.

3. The RBS ATWS analysis demonstrated that for the scenarios in which SLCS injection is initiated in accordance with EPGs, maintaining water level at TAF+5ft or higher would provide sufficient shutdown capability. This finding eliminates the need to lower water level to TAF as proposed by EPG in which case uncertainties in the level reading may potentially introduce adverse impact on the core cooling. In addition, the RBS ATWS analysis shows that no ADS would be required because the core can be adequately shutdown by the SLCS. In the very unlikely case of SLCS failure, ADS would be required no matter where the water level is maintained.

REFERENCES

1. NED0-31331, "Emergency Procedure Guidelines, Revision 4," March 1987

2. NEDE-31096-P-A, Claassen, L. B., et. al. , "Anticipated Transients Without Scram; Response to NRC ATWS Rule IOCFR50 62," February 1987.

3. Jensen, P. J., et. al. , "Anticipated Transients Without Scram Analysis for the River Bend Nuclear Station," Proceedings Fourth International RETRAN Conference, EPRI NP-4558-SR, May 1986, pp 23-1 to 23-21.

4. NEDC-30921, "Assessment of ATWS Compliance Alternatives," March 1985.

5. Motloch, C. G., et. al. , "A Methodology for the Calculation of Boron Concentration in Boiling Water Reactors," Proceedings Third International RETRAN Conference, EPRI NP-3803-SR, February 1985, pp. 21-1 to 21-16.

6. NEDC-22166, "Power Suppression And Boron Remixing Mechanism for General Electric Boiling Water Reactor Emergency Procedures Guidelines," August 1983.

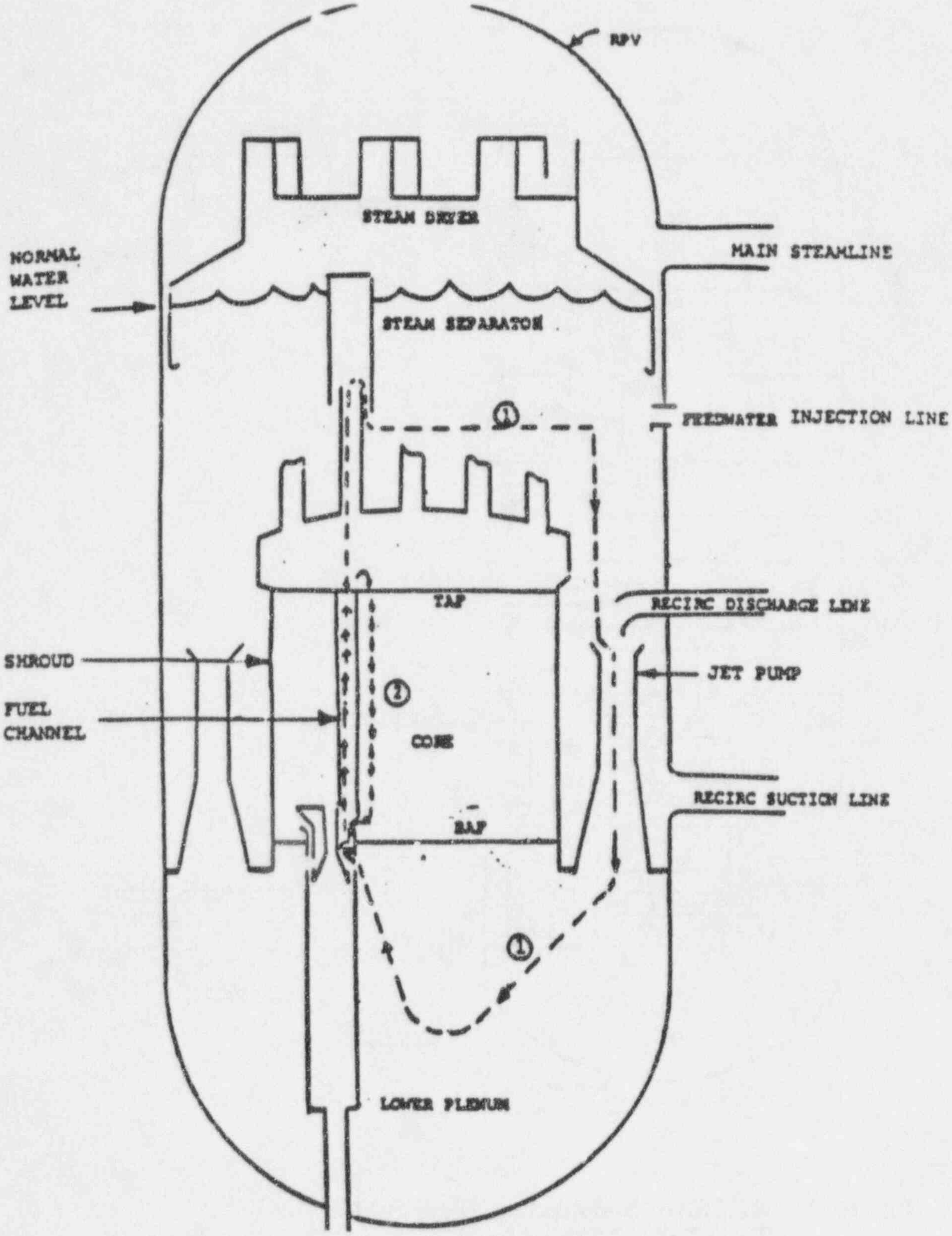


Figure 1. Internal and External Circulation Flow Paths at Low Power Conditions

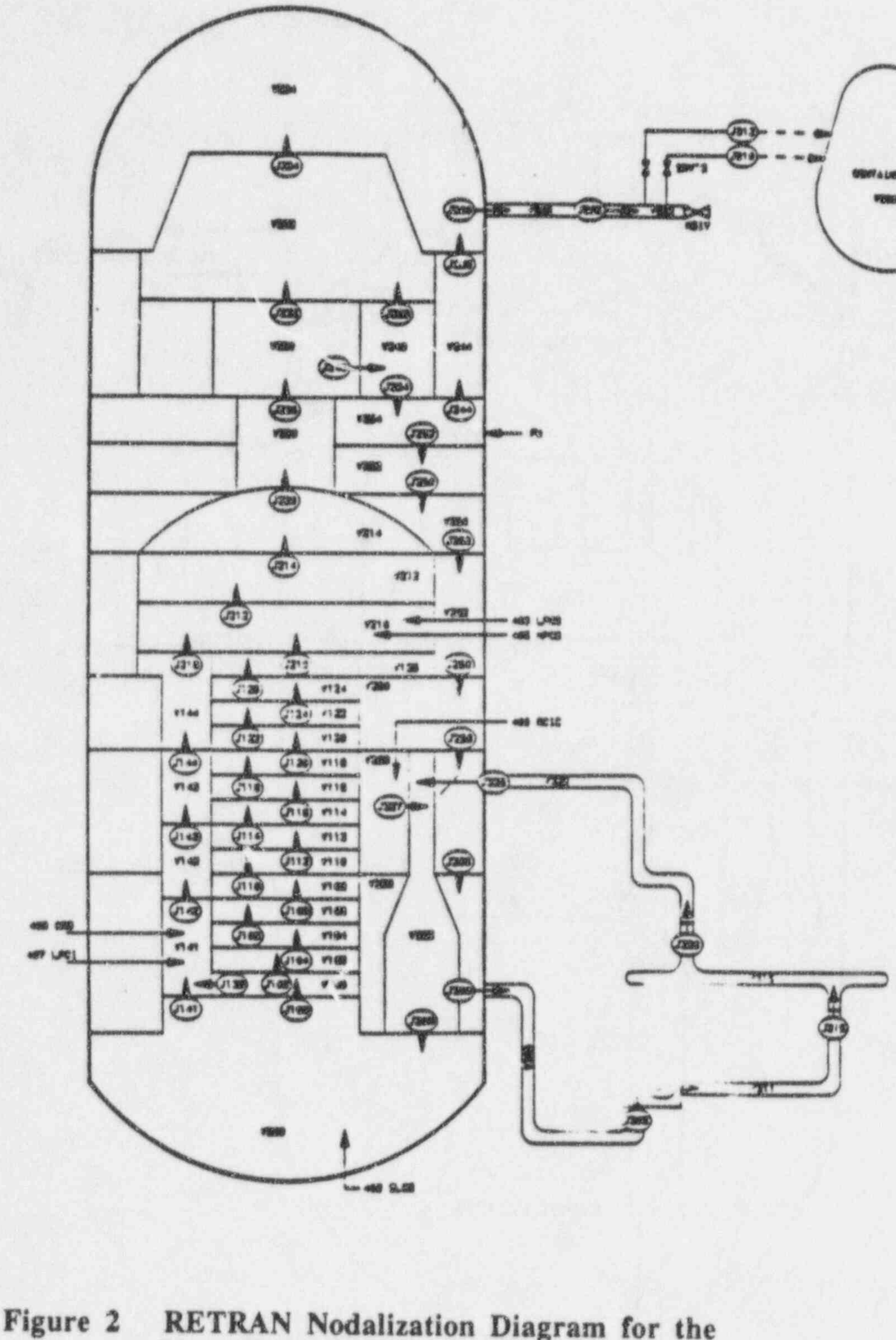


Figure 2 RETRAN Nodalization Diagram for the River Bend ATWS Model

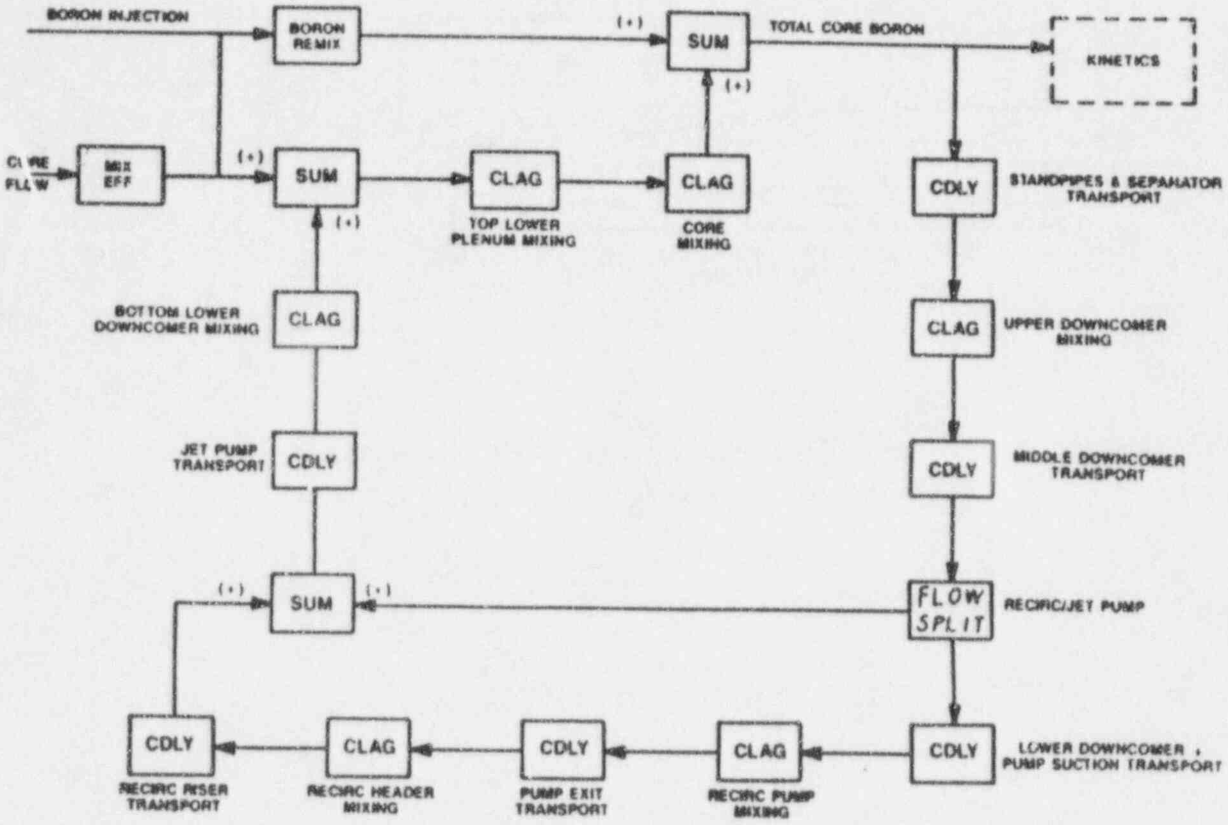


Figure 3. Simplified Boron Concentration Model Schematic

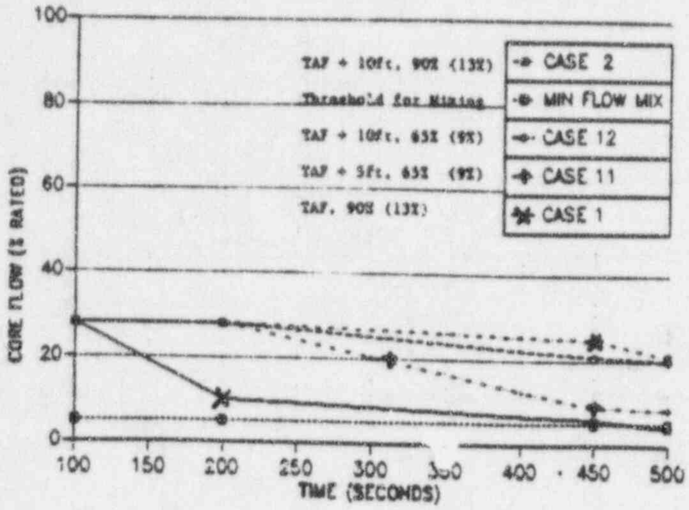


Figure 4. Core Flow Percentage With Respect To Different Water Levels

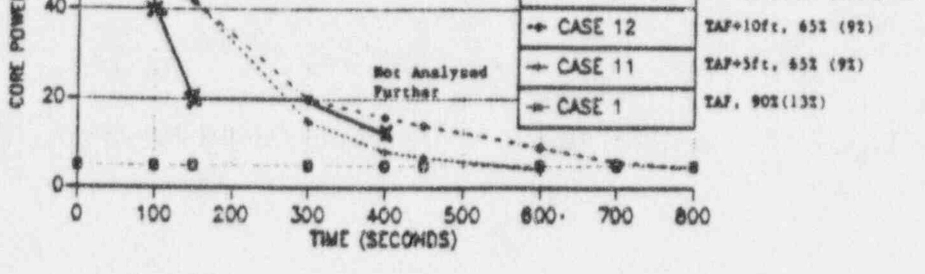


Figure 5. Core Power Percentage With Respect to Different Water Levels

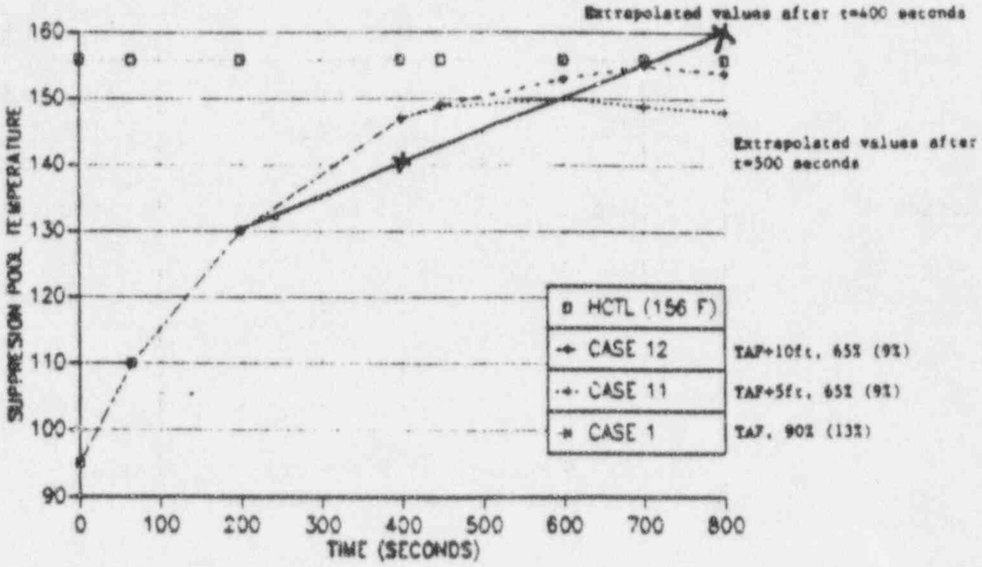


Figure 6. Suppression Pool Temperature With Respect to Different Water Levels

Table 1. A Comparison of Operator Actions Considered in the ATWS Analysis.

	SLCS Delay	Level Reduction Delay	RHR Initiation Delay
Action 1	120 sec	30 sec	600 sec
Action 2	72 sec	90 sec	90 sec

Table 2. A Summary of Boron Tracking Models Used in the ATWS Analysis.

Model 1

- Use an empirical correlation based on 1/6-scale BWR mixing tests
- Correlate mixing coefficient with fraction of core flow and time

Model 2

- Use time dependent LAG and time dependent DELAY
- Model boron flow through a volume as a LAG or DELAY

Model 3

- Ad Hoc Fix of the Model 2
- Boron mass change is restricted to a non-negative value

Model 4

- Use time dependent LAG for all volumes with implicit numerical scheme
- Consider potential flow reversal

Table 3. A Summary of Results and Major Assumptions for Cases 1 through 12.

Case	Water Level	Sodium Pentaborate Concentration	B-10 Enrichment	Operator Actions	Subcooled Injection Model	Boron Tracking Model	Max. SP Temp.
1	TAF	13%	90%	Action 1	Yes	Model 1	160°F
2	TAF+10ft	13%	90%	Action 1	Yes	Model 1	151°F
3	TAF+10ft	13%	40.6%	Action 1	Yes	Model 1	159°F
4	TAF+10ft	13%	40.6%	Action 2	No	Model 1	179°F
5	TAF+10ft	13%	40.6%	Action 2	No	Model 2	167°F
6	TAF+10ft	13%	90%	Action 2	No	Model 2	145°F
7	TAF+5ft	13%	40.6%	Action 2	No	Model 2	160°F*
8	TAF+10ft	13%	65%	Action 2	No	Model 2	152°F*
9	TAF+5ft	13%	65%	Action 2	No	Model 2	155°F*
10	TAF+5ft	9%	65%	Action 2	No	Model 3	142°F
11	TAF+5ft	9%	65%	Action 2	No	Model 4	150°F
12	TAF+10ft	9%	65%	Action 2	No	Model 4	155°F

* Analysis experienced severe numerical oscillations. Maximum suppression pool temperature was obtained based on extrapolation.

Table 4. A Typical Sequence of Events for a MSIV ATWS.

EVENT	TIME (SECONDS)
MSIV Closure	0.01
Vessel Water Level Setpoint Setdown	0.8
Low Level - 4	2.4
Recirc. Pump Trip (High RPV Pressure)*	3.2
SRV Opens	3.1 - 3.9
SP Temp. 95°F	3.5
High Level - 8	4.3
SP Temp. 110°F	62
RHR Initiated	93.5
SLCS Initiated	134 (Power 40%)**
Level Reduction Initiated	152***
First Pass of Boron into the Core	167
Level Reached TAF+5FT	247

* 1145 psia was used in the analysis, actual plant setpoint is 1142 psia.
** Assumes 72-second delay of operator action.
*** Assumes 90-second delay of operator action.

ANALYSIS OF BORON DILUTION IN A FOUR-LOOP PWR

J. G. Sun and W. T. Sha

Energy Technology Division
Argonne National Laboratory
9700 South Cass Avenue
Argonne, IL 60439

Abstract

Thermal mixing and boron dilution in a pressurized water reactor were analyzed with COMMIX codes. The reactor system was the four loop Zion reactor. Two boron dilution scenarios were analyzed. In the first scenario, the plant is in cold shutdown and the reactor coolant system has just been filled after maintenance on the steam generators. To flush the air out of the steam generator tubes, a reactor coolant pump (RCP) is started, with the water in the pump suction line devoid of boron and at the same temperature as the coolant in the system. In the second scenario, the plant is at hot standby and the reactor coolant system has been heated up to operating temperature after a long outage. It is assumed that an RCP is started, with the pump suction line filled with cold unborated water, forcing a slug of diluted coolant down the downcomer and subsequently through the reactor core. The subsequent transient thermal mixing and boron dilution that would occur in the reactor system is simulated for these two scenarios. The reactivity insertion rate and the total reactivity are evaluated.

1 Introduction

A number of mechanisms that have been postulated lead to reactivity–induced transients through boron dilution in a pressurized water reactor (PWR). One mechanism is to quickly pump a slug of cold unborated water through the core, and cause insertion of positive reactivity and thus power excursion and fuel damage. The most conservative assumption in terms of reactivity insertion is that the cold unborated inlet water does not mix with the hot boron–rich water initially in the reactor vessel and the reactor coolant pipes. Therefore, the extent of mixing of cold unborated water with hot boron–rich water is important in realistically quantifying the reactivity insertion due to thermal mixing and boron dilution.

Thermal mixing and boron dilution in the four-loop Zion reactor[1] were analyzed with COMMIX codes.[2–5] COMMIX is a general–purpose, time–dependent, multidimensional computer code for thermal hydraulic analysis of single– or multicomponent engineering systems. It solves a system of conservation equations of continuity, momentum, and energy, and a k–ε two–equation turbulent model. A special feature of the COMMIX code is its porous–media formulation,[6] which represents an unified approach to thermal–hydraulic analysis. A three–dimensional numerical model, based on Cartesian coordinates, was developed for the four–loop reactor system.[7] The major objective of these analyses was to determine the reactivity insertion due to the change of coolant density and boron concentration in the reactor core.

Two boron dilution scenarios were analyzed. In the first scenario, the plant is in cold shutdown and the reactor coolant system has just been filled after maintenance on the steam generators. To flush the air out of the steam generator tubes, a reactor coolant pump (RCP) is started. It is assumed that the coolant in the pump suction line (crossunder line that connects the RCP suction to the outlet of the steam generator) is at the same temperature as the coolant in the system but is devoid of boron. The COMMIX simulation for this scenario is referred to as isothermal–RCP–start calculation.

In the second scenario, it is assumed that all RCPs are recently tripped, and conditions for a restart have been met. The plant is at hot standby and the reactor coolant system has been heated to operating temperature after a long outage. Therefore, natural circulation does not exist because adequate decay heat is lacking. This condition will lead to stagnation of the coolant in the reactor system. It is assumed that boron dilution took place in one of the four pump suction lines during the outrage. When the RCP in the diluted loop is started (with the suction line filled with cold unborated water), a slug of diluted coolant will be forced down the downcomer and subsequently through the reactor core. This simulation is referred to as hot–RCP–start calculation.

In these two analyses, the transient flow, the boron distribution in the first analysis, and the boron and temperature distributions in the second analysis, are computed with the COMMIX code. The reactivity insertion due to the boron dilution and the density change is then estimated.

2 Layout of Four–Loop Zion Reactor

The Zion reactor was selected as the model PWR because it has previously been modeled by COMMIX.[7] It is a typical four–loop PWR with a capacity of 1040 MWe. Figure 1 shows a top view of the plant layout, and Fig. 2 shows a side view of the plant layout. The layout of the internal structures of the Zion reactor vessel is illustrated in Fig. 3.

At normal operation, inlet coolant flow from the cold leg enters the vessel inlet nozzles and proceeds down the annulus between the core barrel and the vessel wall, flows on both sides of the thermal shield, and then into the plenum at the bottom of the vessel. It then turns and flows up through the lower support plate, passes through the intermediate diffuser plate and through the lower core plate. After passing through the core, the coolant enters the area of the upper support structure and then flows generally radially to the outlet nozzles of the core barrel and directly through the vessel outlet nozzles to the hot leg.

A small amount of water also flows between the baffle plates and core barrel. Similarly, a small amount of the entering flow is directed into the vessel head plenum. Both these flows eventually are directed into the upper support structure plenum and exit through the vessel outlet nozzles to the hot leg.

3 The Numerical Model

3.1 Geometry

The numerical model of the four–loop Zion reactor coolant system is an extension of the one–loop model developed in Ref. 1. A schematic layout of the three–dimensional model for one loop is shown in Fig. 4. A top view of the model is shown in Fig. 5. The model contains 20 x 28 horizontal partitions and 20 vertical partitions, for a total of 3334 computational cells.

The vertical partitions of the model shown in Fig. 6 correspond to the major components of the reactor vessel, as illustrated in Fig. 3. From Figs. 3 and 6 it is evident that most of the axial partitions match in a natural way. The axial height of the upper and lower domes of the reactor vessel were adjusted to ensure correct fluid volume. The horizontal dimensions of the grids were determined from the cross–sectional areas of the major components and from the areas between the components.

The steam generators are partitioned to preserve the height and flow areas. All circular pipes, i.e., cold legs, hot legs, and crossunder lines, are modeled with rectangular cross sections. The cross–sectional areas of the modeled pipes are the same as those of the corresponding circular pipes.

To account for the surfaces and volumes occupied by the solid structures in the flow domain, the directional surface porosity and the volume porosity are specified for various components.[1,7] The directional surface porosity used in the COMMIX code is defined by the equation

$$\gamma_{x_i} = \frac{\text{Fluid flow area in direction } x_i}{\text{Total area in direction } x_i},$$

and the volume porosity is defined by

$$\gamma_v = \frac{\text{Fluid volume of a cell}}{\text{Total volume of a cell}}.$$

3.2 Flow resistances

To account for the frictional resistance of the solid structures in the reactor vessel, a set of seven resistance correlations of the form

$$f = a_l \, Re^{b_l} + c_l \qquad \text{(for laminar flow)}$$
$$= a_t \, Re^{b_t} + c_t \qquad \text{(for turbulent flow)}$$

has been implemented in the model. In these correlations, a, b, and c are correlation coefficients, Re is the Reynolds number, and f is the friction factor. These correlations are used to compute frictional resistances in the x, y, and z directions in the reactor core, core bypass, upper plenum, upper plenum bypass, upper core plate, and in steam generators.

3.3 Heat transfer to walls

The core power and heat capacity of both the reactor and pipe walls were not considered in this model. However, they can easily be implemented. We believe these effects are of secondary importance in terms of assessing reactivity insertion due to thermal mixing and boron dilution.

3.4 Fluid physical properties

In the analyses, boron and water are treated as separate components. Because the boron concentration in the system is very small, the thermophysical property of the boron/water mixture is essentially the same as that of unborated water.

3.5 Initial Conditions

For the isothermal–RCP–start calculation, the reactor coolant system was initially filled with borated water at 25°C, 155 bars, and zero velocity. A boron concentration of 2200 ppm, corresponding to a mass fraction of 1.32×10^{-3}, was uniformly distributed in the entire reactor coolant system.

For the hot–RCP–start calculation, the reactor coolant system was initially filled with borated water at 297°C, 155 bars, and zero velocity. A boron concentration of 1200 ppm, corresponding to a mass fraction of 7.209×10^{-4}, was uniformly distributed in the entire reactor coolant system.

3.6 Boundary Conditions

The inlet is at the suction side of the RCP in the second loop, as shown in Fig. 5. The inlet conditions for the two calculations, the isothermal– and the hot–RCP–start, are listed below.

(1) In the isothermal–RCP–start calculation, we assume that a slug of unborated water, initially in the crossunder pipe and half the volume of the RCP, is pumped into the cold leg in Loop 2. The temperature of the inlet water is the same as that in the system. The volume of the slug is 4.87 m^3 (168.9 ft^3), and the increase of pump flow rate with time is shown in Fig. 7.[1] The pump reaches its normal operating flow rate of 130,000 gpm in 25 s. From Fig. 7, it can be determined that the slug of cold unborated water will pass through the pump in 8 s. After 8 s, the boron concentration in the inlet water becomes normal (2200 ppm).

(2) In the hot–RCP–start calculation, the temperature of the inlet slug is low, i.e. 25°C. The unborated slug has the same volume as that in the above calculation. After the slug passes through the pump, boron concentration at the inlet returns to 1200 ppm, and the temperature to 297°C, which is the initial temperature of the system.

It should be noted that in these two calculations, the slug water is assumed only in Loop 2. There is no boron dilution in the other three loops and those RCPs are assumed open for coolant pass through.

The outlet is located at the exit side of the steam generator in the second loop, as shown in Fig. 5. The velocity and temperature gradients at the outlet were set to zero, i.e.,

$$\frac{\partial u_z}{\partial z} = 0, \qquad \frac{\partial T}{\partial z} = 0.$$

Velocity and heat flux were zero on the surfaces of all solid walls.

4 COMMIX Calculational Results

4.1 Boron Dilution under Isothermal–RCP–Start Condition

The transient of the boron dilution in the four–loop reactor coolant system is simulated for 35 s. At 35 s into the transient, most of the slug has been pushed out of the reactor vessel. Figures 8a–d show the velocity distributions in the cold legs (Loops 1 and 2) and reactor vessel at 6, 9, 13, and 18 s into the transient, respectively. At 6 and 9 s into the transient, the inlet velocity is lower. Water in the reactor core and lower plenum is simply pushed upward with nonuniform velocity distribution. At 13 s into the transient, a small recirculation can be observed at the left corner in the lower plenum, Fig. 8c. Flow recirculation commonly occurs at high flow rates, or at high Reynolds numbers. At later times, the magnitude of the recirculating velocity increases, as shown in Fig. 8d for 18 s into the transient. Although not seen in Figs. 8a–d, large flow recirculations are observed in the downcomer. These flow recirculations will improve the boron mixing in the reactor vessel.

The inlet flow is bypassed in the loops. In Loop 2, water flows from the reactor vessel to the hot leg and then to the steam generator. In Loops 1, 3, and 4, however, the flow is reversed. This reversal occurs because only the RCP in Loop 2 is running, and the other loops become bypass routes for the inlet flow. The bypass flow reduces the volume of the slug flowing through the core and, therefore, mitigates the boron dilution in the core. The total bypass flow rate accounts for 20.6% of the inlet flow rate. It should be pointed out that the bypass flow rates depend on the modeling of the flow resistance in the steam generator and the pipes.

The inlet unborated water mixes with the boron–rich coolant in the system. The calculational result indicates that little boron mixing occurs in the cold leg. However, strong boron mixing occurs in the downcomer. For instance, at 14 s into the transient, the lowest boron concentration in the lower plenum is 1849 ppm. Therefore, after the downcomer, the boron concentration in the slug has recovered to ≈84% of the initial 2200–ppm boron concentration of the system. The strong mixing is due partially to the large volume of the downcomer and partially to the flow circulations in the downcomer. Additional mixing takes place in the lower plenum and in the reactor core, resulting in an increase of the minimum boron concentration to 2006 ppm in the core at 20 s into the transient. The travel of the slug through the core is shown in Figs. 9a–d, which show that the slug enters the core at ≈16 s and exits the core at ≈22 s into the transient. The figures also show that the boron concentration is not uniform at each level in the core.

Although the boron concentration is not uniform, a mean boron concentration at each horizontal level is helpful in estimating the overall boron dilution in the reactor vessel. The mean boron concentration as a function of vessel height z is shown in Fig. 10 for the transient at 16, 18, 20, and 22 s. Figure 10 also identifies the location of the slug and shows that the mean boron concentration in the core is always higher than 2046 ppm. Therefore, the mean boron concentration in the core is reduced by <154 ppm, or 7%, which indicates moderate mixing in the downcomer and lower plenum. In the most conservative estimation, by assuming no mixing, the boron concentration in the core would have been reduced to zero when the slug entered the core.

Let us denote the average boron concentration in the core as B, the mean rate of boron concentration change ($\Delta B/\Delta t$) can be obtained from Fig. 10. The reactivity insertion rate due to boron concentration change $\Delta K_B/\Delta t$ as a function of time is obtained by multiplying $\Delta B/\Delta t$ with the reactivity coefficient $\partial K_B/\partial B$. For the condition of this analysis, $\partial K_B/\partial B$ is -10.2×10^{-5} ppm^{-1}.[1] The variation of the reactivity insertion rate $\Delta K/\Delta t$ (= $\Delta K_B/\Delta t$) with time is plotted in Fig. 11. It is seen that the reactivity insertion rate in the core increases rapidly after 12 s into the transient and reaches a maximum of ≈0.0034 s^{-1} (1 pcm = 10^{-5} s^{-1}), or ≈\$0.76/s with the conversion factor $\beta = 0.0045$, at 17.5 s into the transient. The total reactivity K is the integration of $\Delta K/\Delta t$ with time and it is also plotted in Fig. 11. The maximum mean reactivity is ≈0.014 (or ≈\$3.1), which is high. Because of the nonuniformity of the boron distribution, the maximum local reactivity rate and reactivity are even higher. The local reactivity rate and reactivity represent the local power density change (or local temperature change with respect to time) and local power density (or local temperature), respectively.

4.2 Thermal Mixing and Boron Dilution under Hot-RCP-Start Condition

The transient of the thermal mixing and boron dilution in the reactor coolant system is simulated for 35 s. The overall velocity distributions are similar to those obtained in the isothermal calculation at corresponding times. However, because the temperature of the slug in this calculation is much lower than that of the water in the reactor vessel, local velocities change because of the buoyancy effect. The velocity of the local flow in the downcomer is greater. As a result, the cold unborated inlet slug will not mix well in the downcomer and in the lower plenum. The slug flows horizontally to the other side in the lower plenum and then turns upward. The total flow bypassed through Loops 1, 3, and 4 is 20.5% of the inlet flow in Loop 2.

Calculated temperature contours are plotted in Figs. 12a–d for 16, 18, 20, and 22 s into the transient. Near the slug front, the isotherms are approximately horizontal, as observed in Figs. 12a and 12b, indicating good mixing of the front portion of the slug in the lower plenum. Before the remainder of the slug is mixed, the larger inlet flow pushes it to the right corner in the lower plenum, as shown in Fig. 12a. The slug then rises at the right side, as shown in Figs. 12b–d.

The calculated distributions of boron concentration are similar to those of the temperature shown in Figs. 12a–d. The slug does not mix well with the vessel water in the downcomer because of the buoyancy effect, and in the lower plenum because of the short residence time. The minimum boron concentration in the lower plenum at 14 s into the transient is 902 ppm, ≈75% of the initial boron concentration, compared with 84% in the isothermal calculation. The mean boron concentration as a function of vessel height z is shown in Fig. 13 for the transient at 16, 18, 20, and 22 s. The mean boron concentration in the core is always >1025 ppm. The maximum decrease of the mean boron concentration in the core is 14.6%, compared with 7% in the isothermal calculation. Or, from the view point of boron mixing of the slug that was initially unborated, its boron concentration reached only to 85.4% of the initial boron concentration of the system in this calculation, whereas, in the isothermal case, it reached 93%. Therefore, the boron mixing for the slug in this calculation is worse than that in the isothermal calculation.

The total rate of reactivity insertion ($\Delta K/\Delta t$) is the sum of that due to the change in boron concentration ($\Delta K_B/\Delta t$) and that due to change in coolant density ($\Delta K_\rho/\Delta t$). The reactivity rate due to boron concentration change $\Delta K_B/\Delta t$ is obtained by multiplying the reactivity coefficient $\partial K_B/\partial B$ by $\Delta B/\Delta t$. The reactivity rate due to density change $\Delta K_\rho/\Delta t$ is the product of density change per second ($\Delta\rho/\Delta t$) multiplied by the reactivity coefficient due to density change ($\partial K_\rho/\partial\rho$). For the condition of this analysis, $\partial K_\rho/\partial\rho = 31.15\text{x}10^{-5}$ m^3/kg (or 0.00499 ft^3/lb).[1] The rate of change in coolant density ($\Delta\rho/\Delta t$) can be obtained by multiplying $\Delta T/\Delta t$ by $\partial\rho/\partial T$, which is -1.61 kg/m^3/°C for water. The variation of the total reactivity insertion rate in the core with time $\Delta K/\Delta t$ is plotted in Fig. 14. It is seen that the reactivity insertion rate in the core increases rapidly after 12 s into the transient and reaches a maximum of ≈ 0.0082 s^{-1} (or $\approx$\$1.8/s) at 15.5 s into the transient. The variation of total reactivity K with time is also plotted in Fig. 14. The maximum reactivity is ≈ 0.029 (or $\approx$\$6.4) at 18 s into the transient, which is very high. Again, because of the nonuniformity of boron distribution, the maximum local reactivity rate and reactivity are even higher.

5 Discussion and Conclusions

1. Transient calculations for thermal mixing and boron dilution in a four-loop PWR coolant system were performed with the COMMIX code. In the isothermal-RCP-start calculation, large flow recirculations were found in the downcomer and small flow recirculations were found in the lower plenum of the reactor vessel. These flow recirculations improved the boron mixing of the unborated inlet slug with the coolant in the reactor vessel. As a result, there was strong boron mixing in the downcomer and moderate boron mixing in the lower plenum and reactor core. As the slug passed through the downcomer, its boron concentration recovered from 0 to 84% of the system boron concentration. Additional boron mixing took place in the lower plenum and reactor core. Furthermore, the bypass of the inlet flow through the other three loops (reverse flows) reduced the volume of the slug passing through the core. The total bypass accounted for 20.6% of the inlet flow rate. The mean reactivity insertion rate ($\Delta K/\Delta t$) increased quickly after 12 s into the transient and reached a maximum of ≈ 0.0034 s^{-1}, or $\approx$\$0.76/s, at 17.5 s into the transient. The maximum reactivity K was ≈ 0.014 (or $\approx$\$3.1) at 20 s into the transient, which is high.

2. In the hot-RCP-start calculation, boron mixing was not as good as that in the isothermal calculation. Although large flow recirculations existed in the downcomer, the flow was dominantly downward as the cold slug was passing through the downcomer, which was due to the buoyancy effect. As the slug passed through the downcomer, its boron concentration recovered from 0 to 75% of the system boron. It was also found that the cold slug does not have enough time to mix with the vessel coolant in the lower plenum. It was pushed to the opposite side of the inlet loop by the large inlet flow rate. Moderate thermal and boron mixing occurs in the lower plenum and in the reactor core. The total bypass accounted to 20.5% of the inlet flow rate. The mean reactivity insertion rate ($\Delta K/\Delta t$) due to both the coolant density change and boron concentration change also increased quickly after 12 s into the transient and reached a maximum of ≈ 0.0082 s^{-1}, or $\approx$\$1.8/s, at 15.5 s into the transient. The maximum reactivity K was ≈ 0.029, or $\approx$\$6.4, at 18 s into the transient, which is very high.

3. The results presented here appear reasonable. However, this was the first time we used the COMMIX code to predict boron concentration distribution. We are interested to validate our model

with experimental boron dilution data.

4. The proper procedure to compute reactivity feedback due to boron dilution and thermal mixing is to perform an integral, time–dependent, three–dimensional, thermal–hydraulic–neutronic calculation. The present calculations are limited to time–dependent, three–dimensional, thermal–hydraulic calculations. Reactivity feedback due to boron dilution and thermal mixing is estimated via reactivity coefficients[1] with respect to change in boron concentration ($\partial K/\partial B$) and coolant density ($\partial K/\partial \rho$). These reactivity coefficients were very conservative values based on point kinetics.

5. The complex geometry of the four–loop Zion reactor has been greatly simplified and represented in Cartesian coordinates. Because a porous medium formulation is used in the COMMIX code, both volume and area occupied by fluid in each reactor component are correctly simulated via volume porosity and directional surface porosities. These simplification in geometrical representation in our numerical model will affect the flow field.

Acknowledgments

The authors gratefully acknowledge the encouragement and support of Gene Trager, George Lanik, and Jack Rosenthal of the Office for Analysis and Evaluation of Operational Data, U.S. Nuclear Regulatory Commission, and thank K. Ramsden, D. Redden, and S. Ahmend of Commonwealth Edison Company for useful discussions.

References

1. J. G. Sun, and W. T. Sha, *Analysis of Thermal Mixing and Boron Dilution in a PWR*, NUREG/CR–5822, Argonne National Laboratory Report ANL–91/43 (1993).
2. W. T. Sha, H. M. Domanus, R. C. Schmitt, J. J. Oras, and E. I. H. Lin, *COMMIX–1: A Three–Dimensional Transient Single–Phase Computer Program for Thermal–Hydraulic Analysis*, NUREG/CR–0785, Argonne National Laboratory Report ANL–77–96 (Sept. 1978).
3. H. M. Domanus, R. C. Schmitt, W. T. Sha, and V. L. Shah, *COMMIX–1A: A Three–Dimensional Transient Single–Phase Computer Program for Thermal–Hydraulic Analysis of Single and Multicomponent Systems. Vol. I User's Manual*, and *Vol. II Assessment and Verification*, NUREG/CR–2896, Argonne National Laboratory Report ANL–82–25 (Dec. 1983).
4. F. F. Chen, H. M. Domanus, C. C. Miao, R. C. Schmitt, V. L. Shah, and W. T. Sha, *COMMIX–1B: A Three–Dimensional Transient Single–Phase Computer Program for Thermal–Hydraulic Analysis of Single and Multicomponent Systems: Vol. I Equations and Numerics*, and *Vol. II User's Manual*, NUREG/CR–4348, Argonne National Laboratory Report ANL–85–42 (Sept. 1985).

5. H. M. Domanus, Y. S. Cha, T. H. Chien, R. C. Schmitt, and W. T. Sha, *COMMIX–1C: A Three–Dimensional Transient Single–Phase Computer Program for Thermal–Hydraulic Analysis of Single and Multicomponent Engineering Systems: Vol. I Equations and Numerics*, and *Vol. II User's Manual*, NUREG/CR–5649, Argonne National Laboratory Report ANL–90/33 (Nov. 1990).

6. W. T. Sha, B. T. Chao, and S. L. Soo, *Time Averaging of Local Volume–Averaged Conservation Equations of Multiphase Flow*, NUREG/CR–3434, Argonne National Laboratory Report ANL–83-49 (July 1983).

7. V. L. Shah, *Thermal Hydraulic Simulation of Zion Power Plant under Normal Operating Conditions*, Argonne National Laboratory Report ANL/ATHRP-36 (1985).

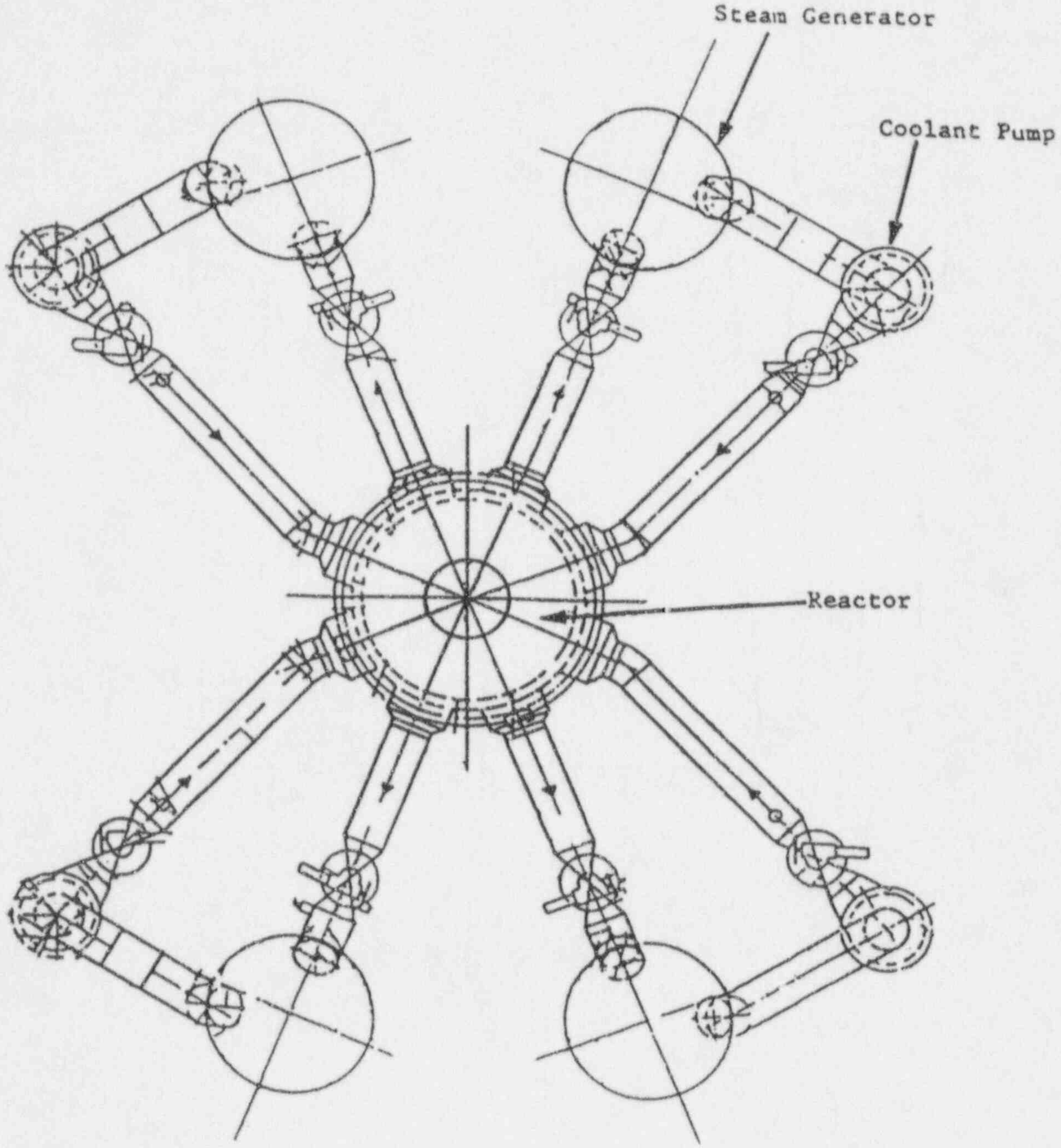


Fig. 1. Top view of layout of Zion reactor coolant loops

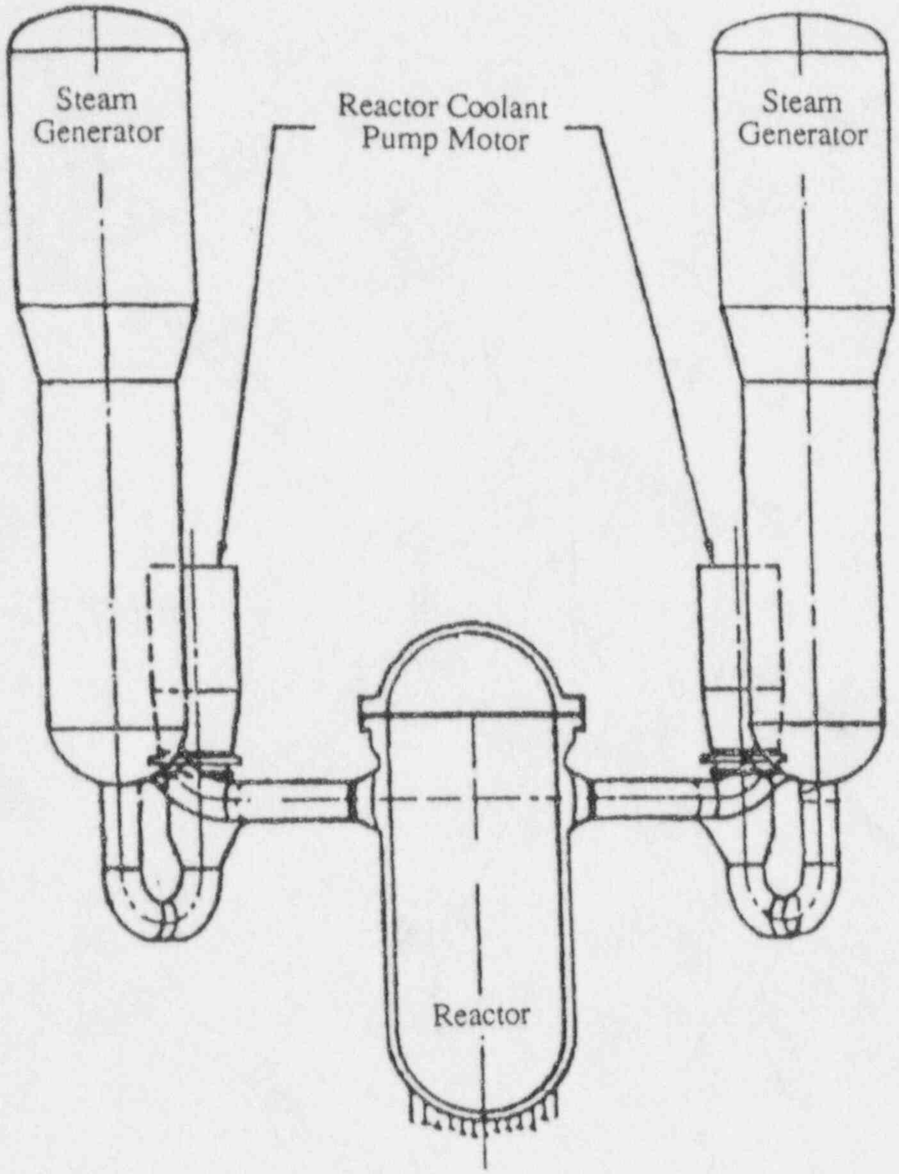


Fig. 2. Side view of layout of Zion reactor coolant loops

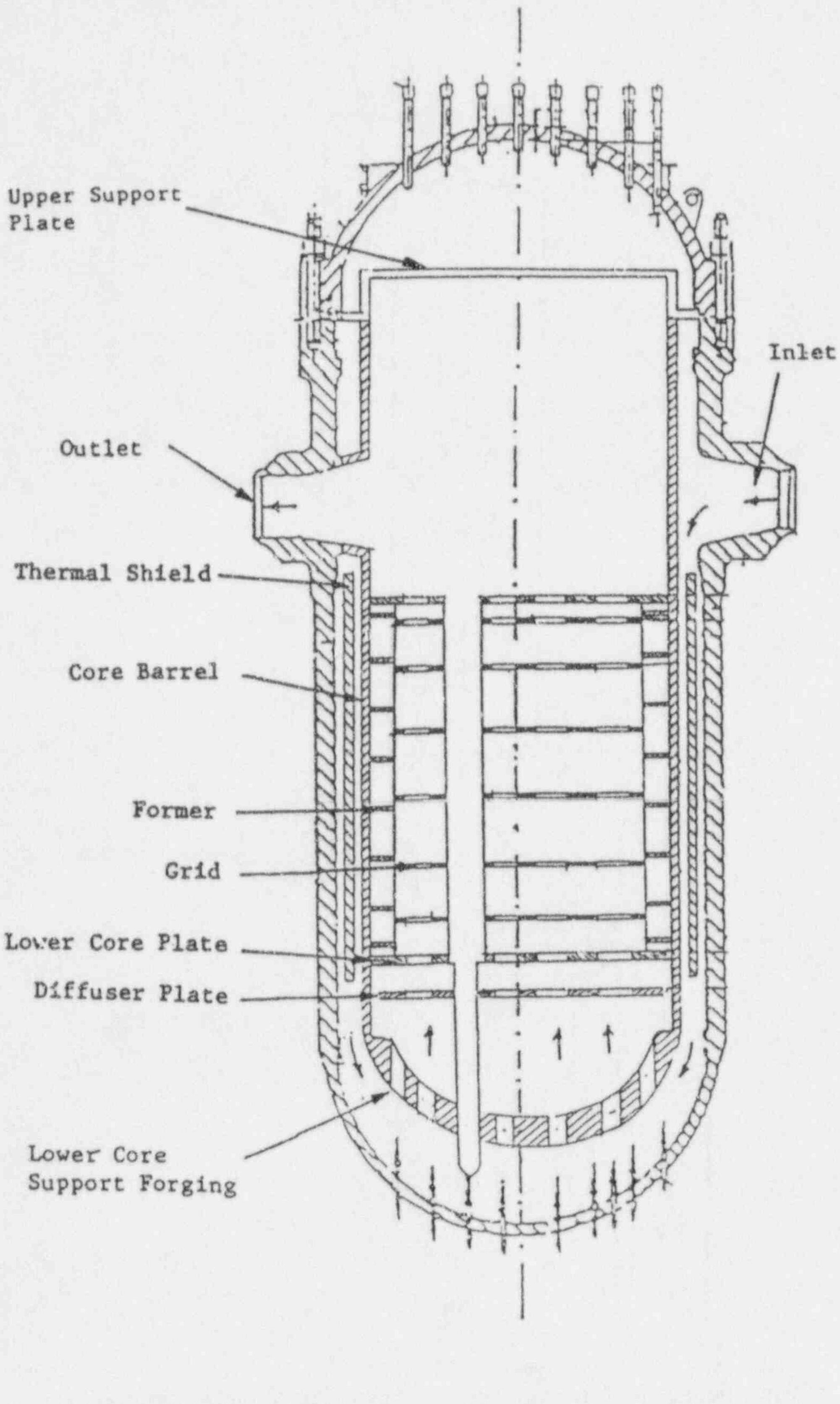


Fig. 3. Schematic layout of internal structures of Zion reactor

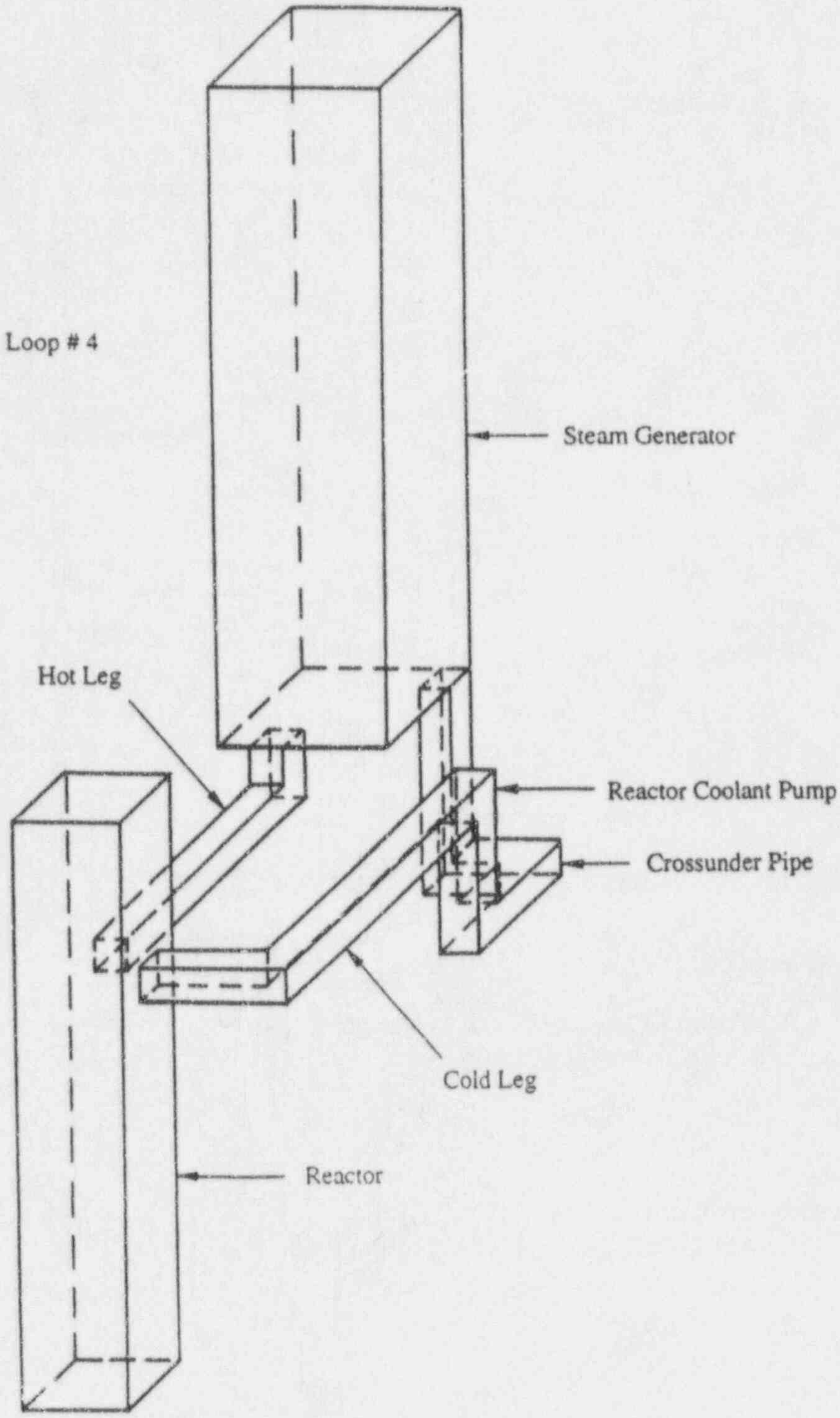


Fig. 4. Schematic layout of three–dimensional model for Zion reactor Loop 4

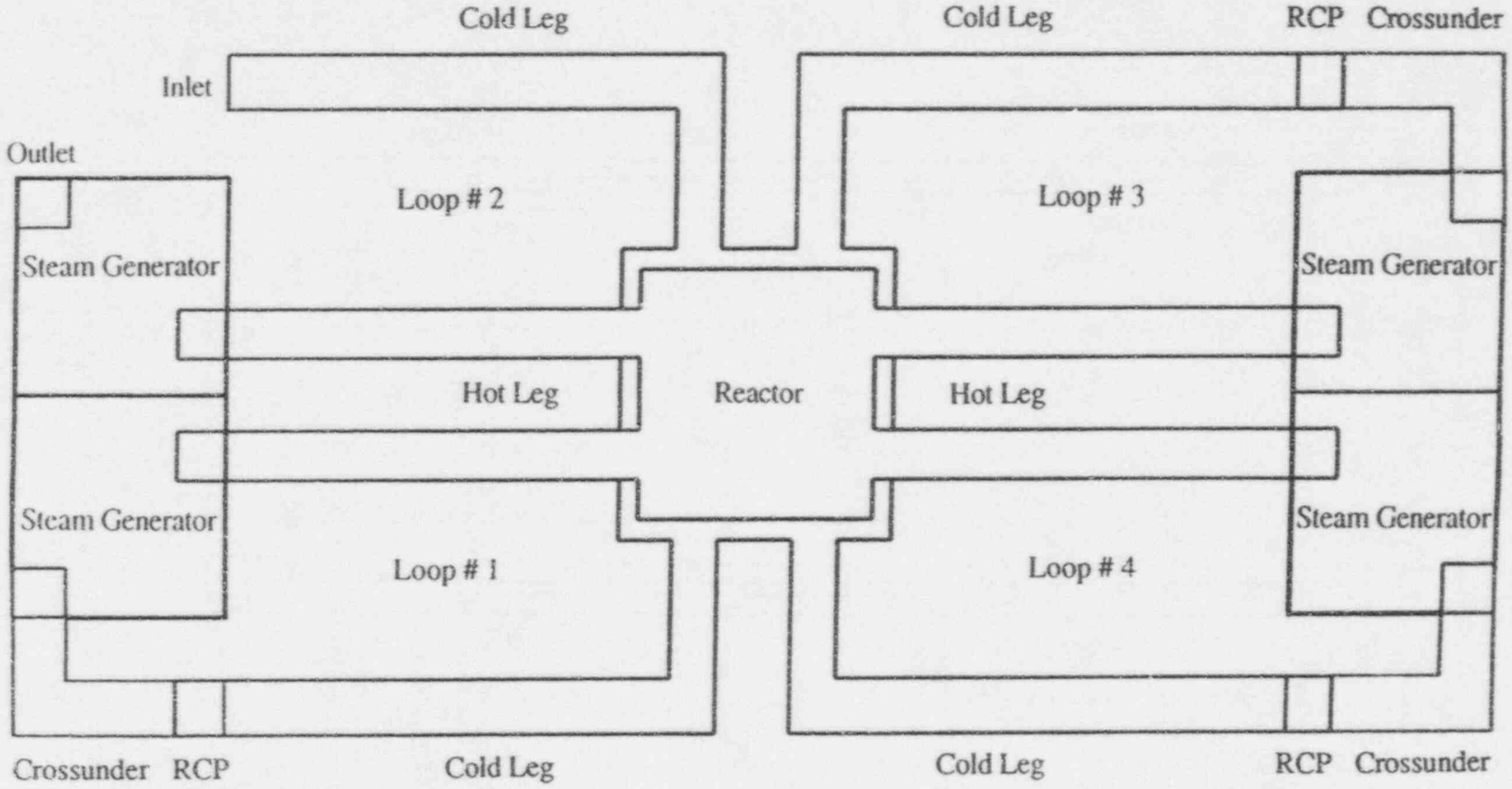


Fig. 5. Top view of numerical model of Zion reactor coolant loops

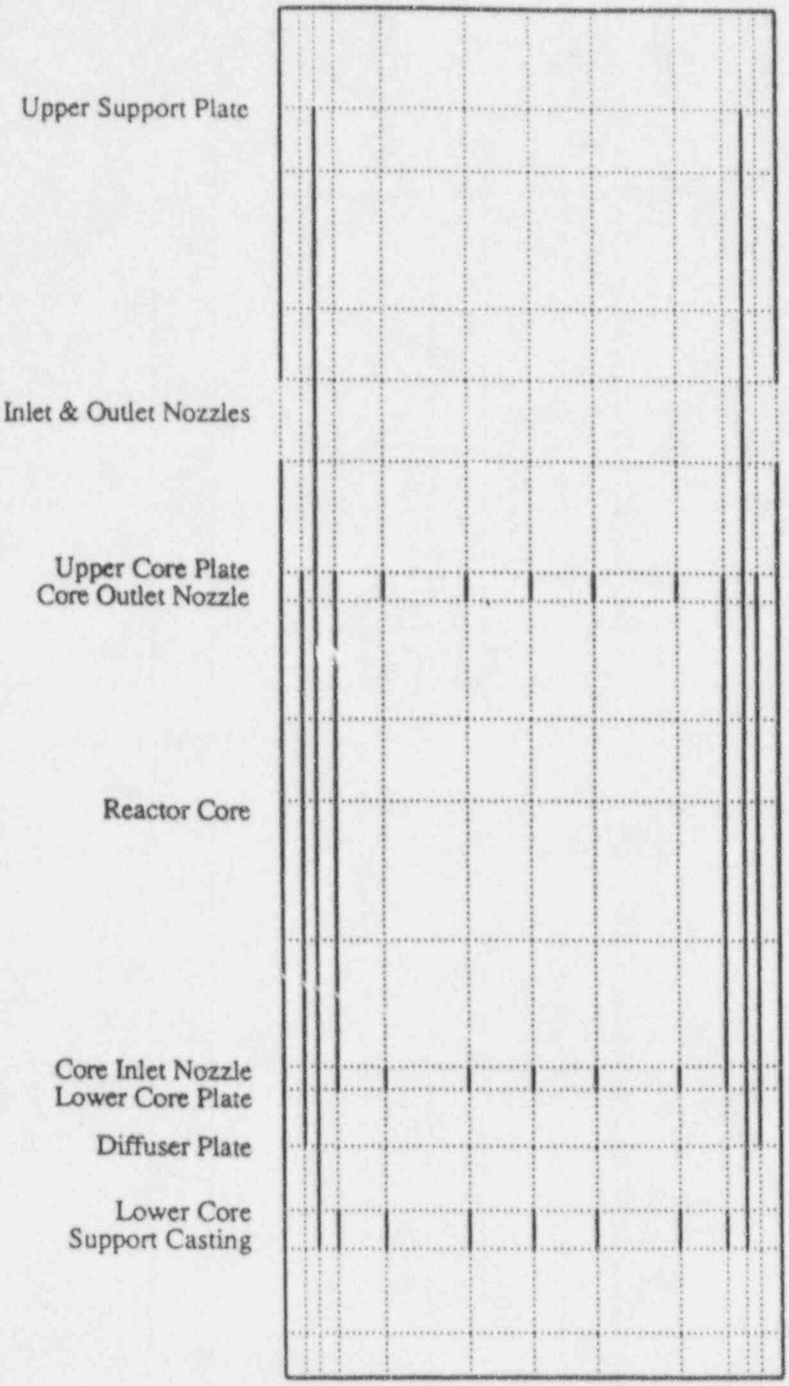


Fig. 6. Vertical partitions of reactor vessel

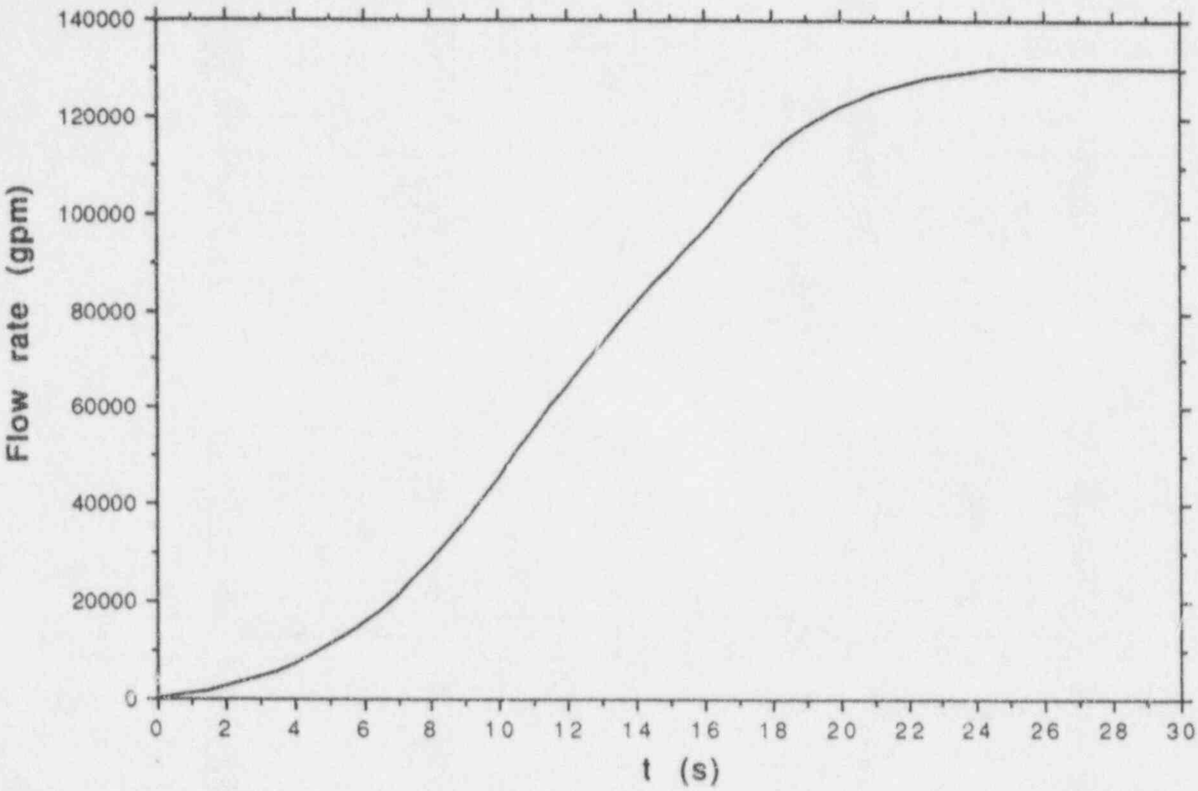


Fig. 7. Increase in flow rate with time at start of RCP

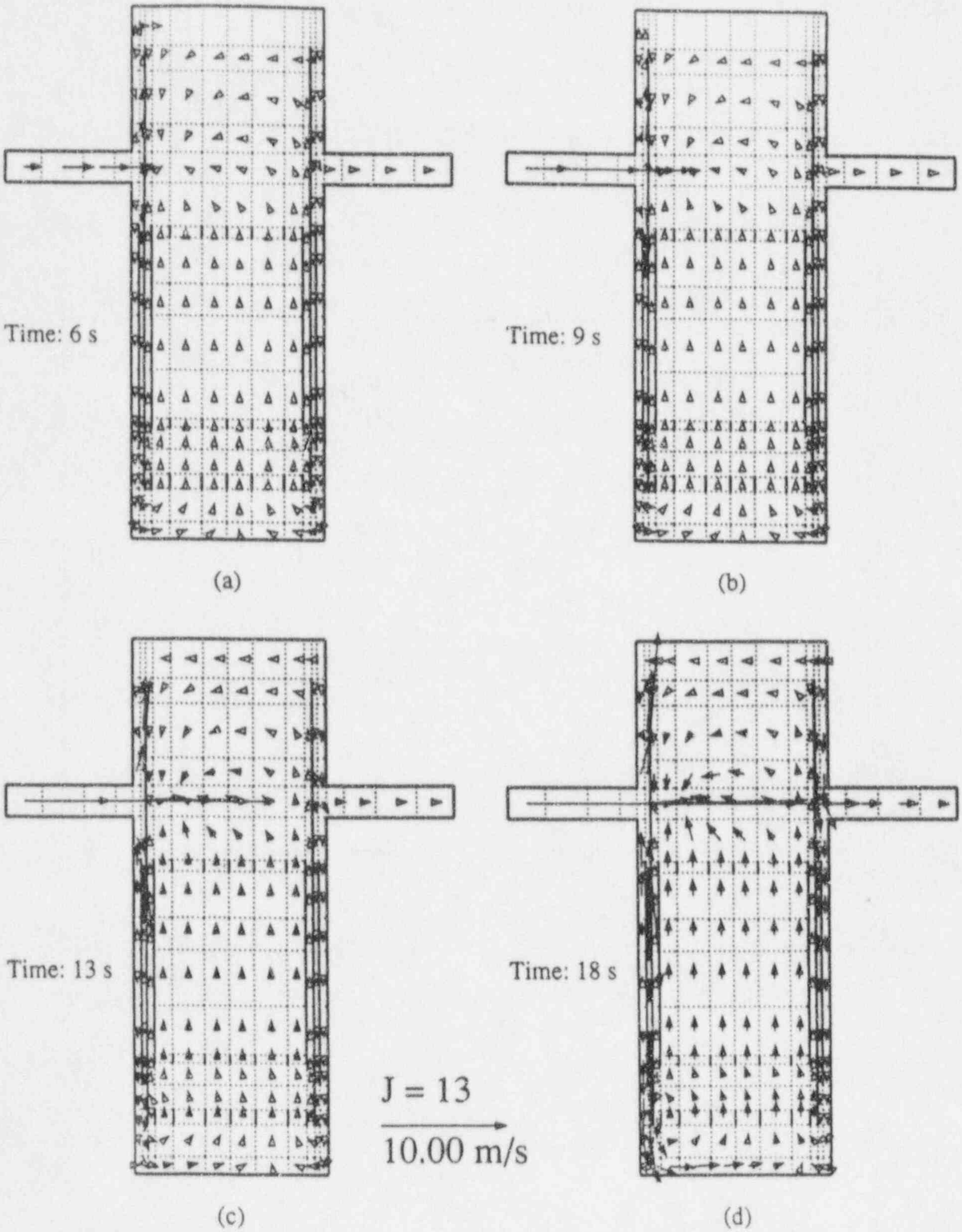


Fig. 8. Velocity distributions in cold legs and reactor vessel for isothermal–RCP–start calculation

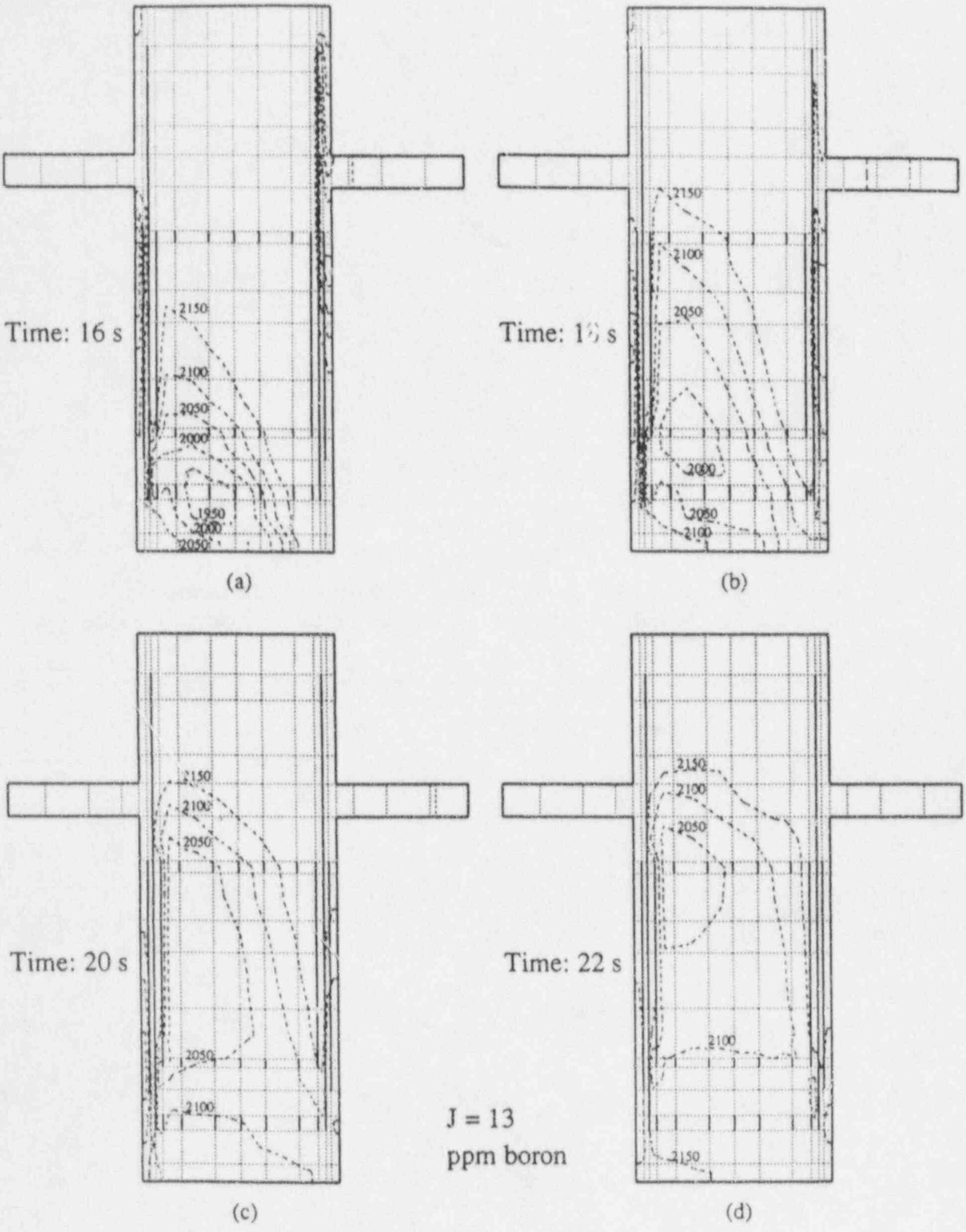


Fig. 9. Contours of the boron concentration in reactor vessel for isothermal–RCP–start calculation

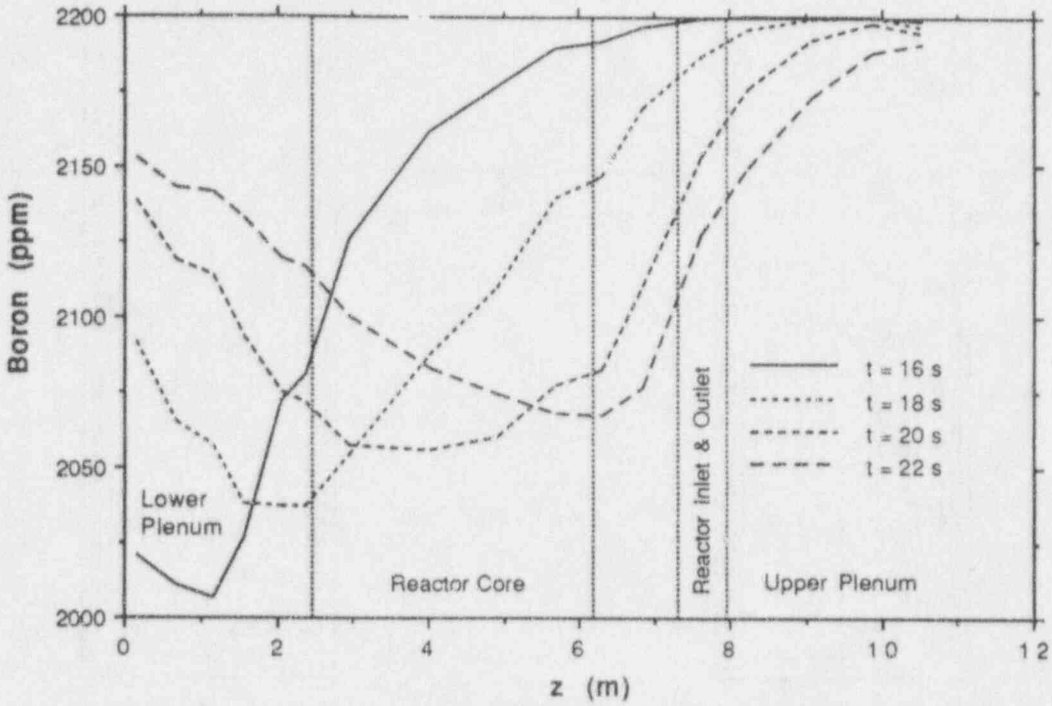


Fig. 10. Variation of boron concentration along axis of reactor vessel at 16, 18, 20, and 22 s into transient for isothermal–RCP–start calculation

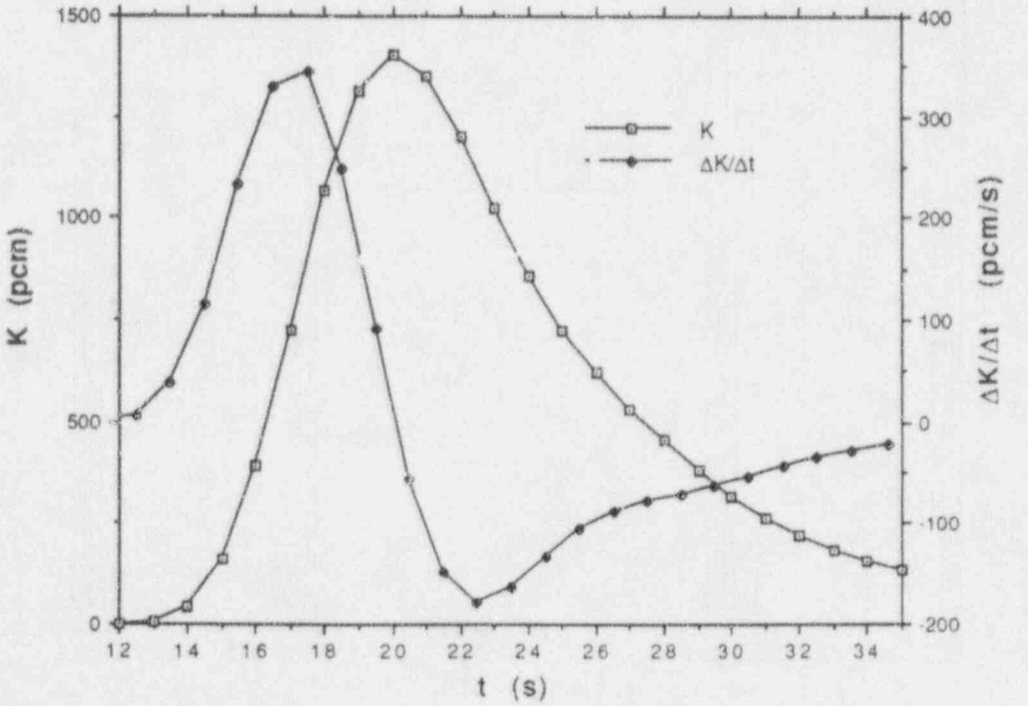


Fig. 11. Variation of mean reactivity and reactivity insertion rate with time in reactor core for isothermal–RCP–start calculation

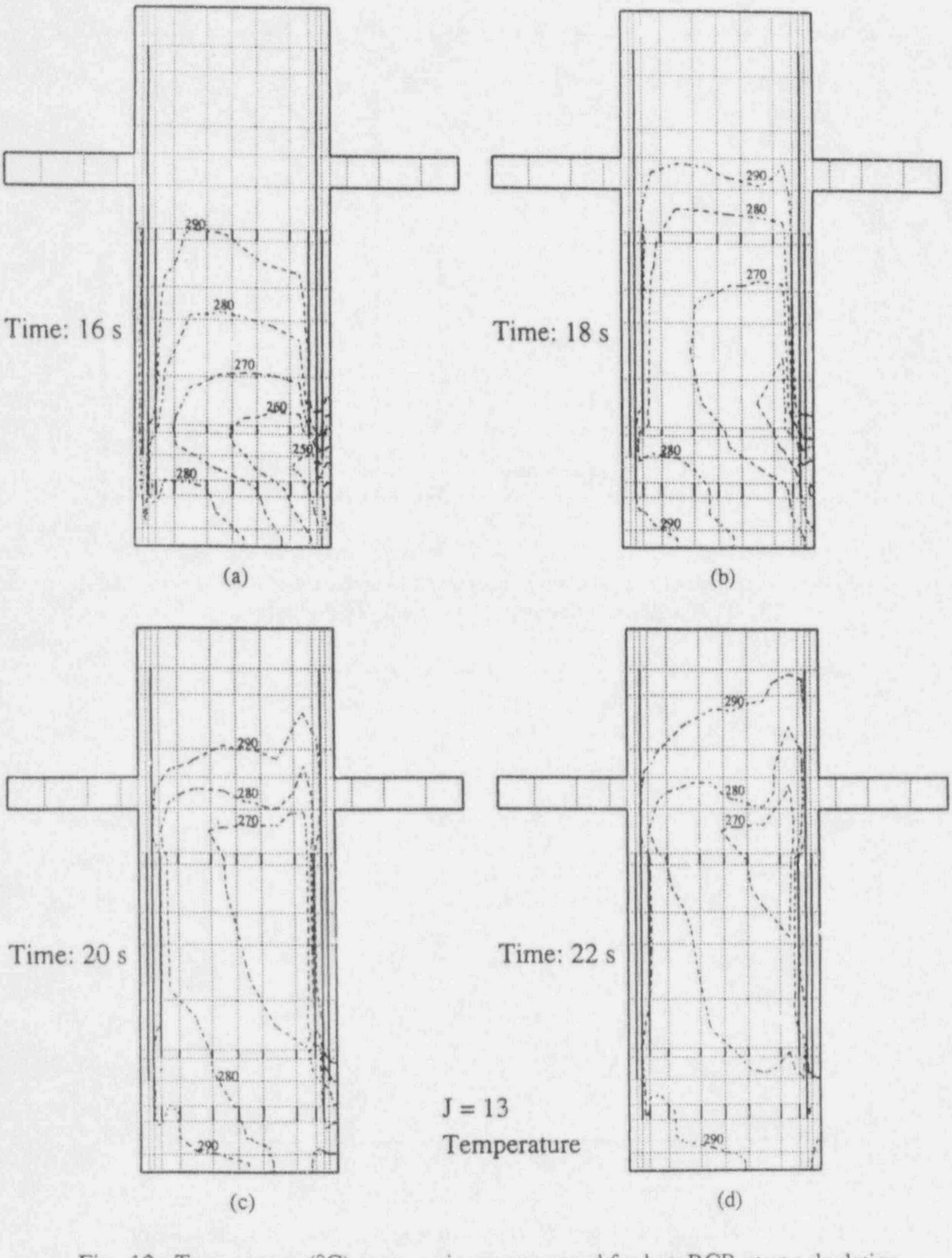


Fig. 12. Temperature (°C) contours in reactor vessel for hot–RCP–start calculation

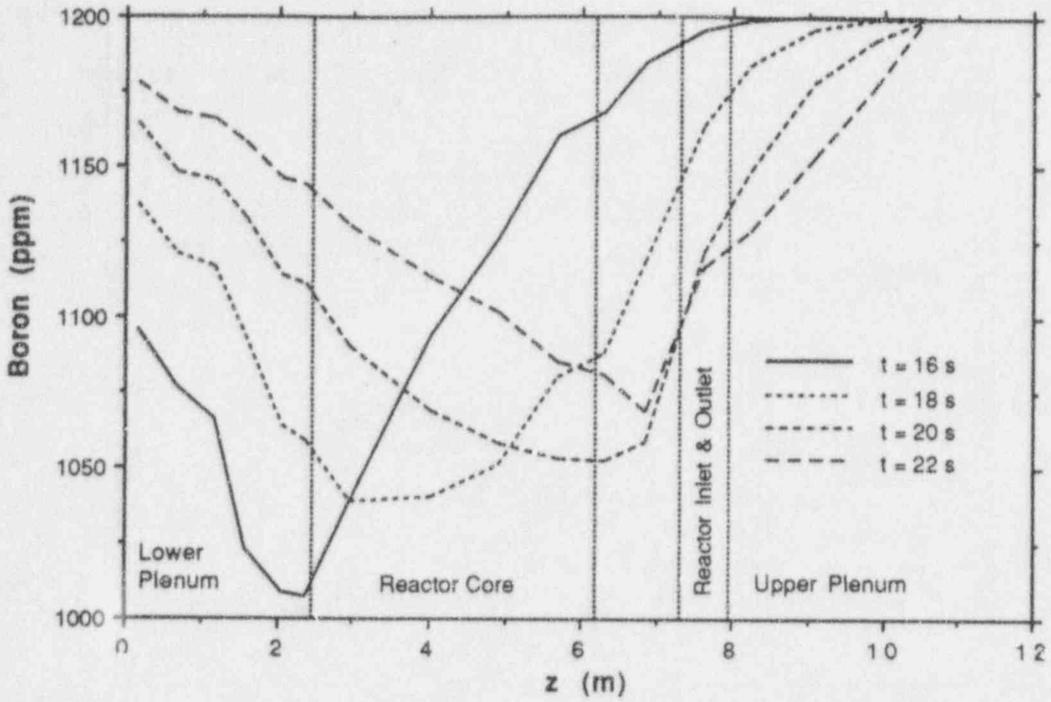


Fig. 13. Variation of mean boron concentration along axis of reactor vessel at 16, 18, 20, and 22 s into the transient for hot–RCP–start calculation

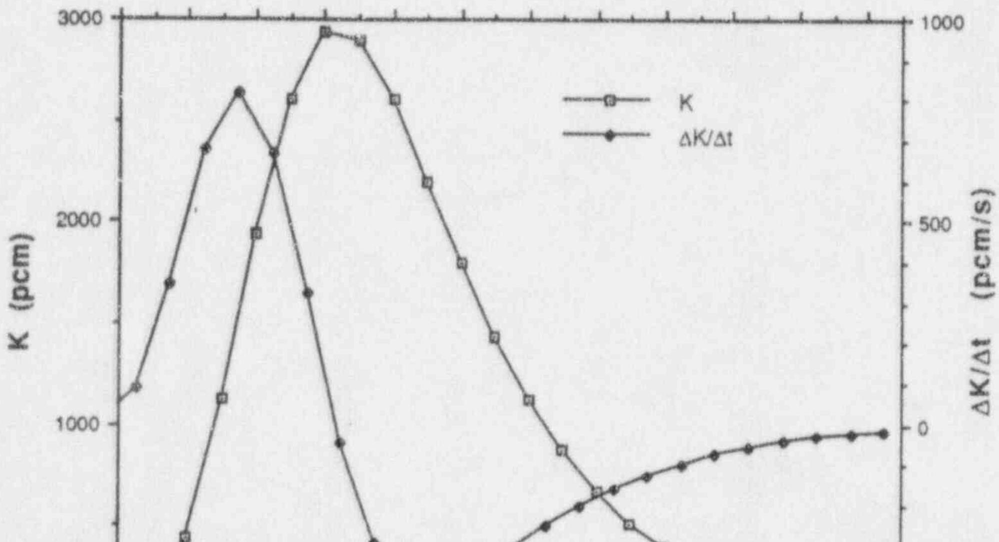


Fig. 14. Variation of mean reactivity and reactivity insertion rate with time in reactor core for hot–RCP–start calculation

KEMA

FLUENT and TRAC-PF1 Analyses of Boron Dilution

W.J.M. de Kruijf,
KEMA, P.O. Box 9035
6800 ET Arnhem, The Netherlands

SUMMARY

In this paper a chronological overview is presented of the work done at KEMA on boron dilution in a Pressurized Water Reactor (PWR). This work has been focused on calculating boron dilution when an unborated water slug enters the reactor vessel at a pump start-up. Two different thermal-hydraulic codes have been used for this purpose: FLUENT and TRAC-PF1. FLUENT is a computational fluid dynamics code, while TRAC-PF1 is specifically written to analyse transients in a pressurized water reactor. This paper gives an idea of the advantages and the disadvantages of these two codes and shows some comparisons between the results. Both codes appear to provide reasonable results for scoping analyses purposes and show that prompt criticality would require a substantial volume of unborated water feed. A phenomenon which appeared to be very important is numerical diffusion. For plant specific quantitative anlayses it is recommended to improve TRAC-PF1 to better model the boron dilution (reduce numerical diffusion, model turbulent diffusion). A detailed comparison between TRAC-PF1 and a code as FLUENT for a relatively simple case would enable a quantitative verification of the improved TRAC-PF1 model, and would provide confidence to use TRAC-PF1 for more complicated transients.

1 INTRODUCTION

Boron dilution is a possible mechanism for a reactivity initiated accident in a PWR. Therefore, it has received a lot of attention in recent years. The work at KEMA (KEMA, 1991; KEMA, 1992; KEMA, 1994a; KEMA, 1994b; KEMA, 1994c) has been focused on the calculation of the amount of mixing of an unborated water slug entering the reactor vessel at a pump start-up with the borated water already present in the reactor vessel. This amount of mixing is important in view of the strong dependence of the reactivity of the core on the boron distribution.

The calculations have all been done for a typical 2-loop reactor with a thermal power of about 1350 MW. Figure 1 shows a schematic drawing of one loop of the primary system.

The calculations were accomplished in two consecutive steps. First, thermal hydraulic calculations have been done to determine diluted boron concentrations in the reactor vessel, using respectively the computational fluid dynamics code FLUENT (FLUENT, 1989) and the transient analysis code TRAC-PF1 (LANL, 1992). Secondly, k-effective calculations have been

file ref. 40904-NUC 95-2738

done to determine the multiplication factor (k-effective) of the core. These neutronics calculations have been done with LWRSIM (KEMA, 1994d), which uses the boron distribution calculated in the first step. Since the work has been focused on the calculation of the boron dilution, the k-effective results as they have been calculated by LWRSIM are merely stated.

Several scenarios for boron dilution have been studied with the code FLUENT. These scenarios consider a cold core with all rods in. The initial boron concentration is 2200 parts per million (ppm). It is assumed that the main circulation pump of one loop of the primary system will be started according to its start-up curve. The scenarios differ in the amounts of unborated water that are pumped into the core. It should be stressed that these scenarios are merely hypothetical. The amounts of unborated water considered reflect the volume of the cold leg of the primary loop, the volume of the pump housing, and the volume of the pipe between steam generator and pump. These volumes are about 5 m^3 each. So, the following scenarios have been considered:

scenario 0: continuous supply of unborated water
scenario 1: 5 m^3 unborated water is pumped into the vessel,
followed by a continuous supply of borated water (2200 ppm).
scenario 2: as scenario 1, but now with 10 m^3.
scenario 3: as scenario 1, but now with 15 m^3.

Basic results of calculations on these scenarios with version 3.02 of FLUENT are presented in section 2 (KEMA, 1991). One specific scenario (3) has been picked to study the effects of numerical diffusion with FLUENT. Results of this sensitivity analysis are presented in section 3 (KEMA, 1994a). Because FLUENT version 3.02 is not very suited to model the entire primary system of the reactor and the detailed structures in the core, the same specific scenario (3) has also been studied with TRAC-PF1. TRAC-PF1 is specifically dedicated to the calculation of transients in a pressurized water reactor. Section 4 deals with these calculations (KEMA, 1994b). In section 5 the basic results of a study with TRAC-PF1 on the vessel nodalization are presented (KEMA, 1994c). Again, scenario 3 is used for these calculations. An overview of the calculations which are discussed in this paper is given in table 1. At the end of this paper some general conclusions and recommendations for further research on the matter at hand are given.

2 FLUENT CALCULATIONS ON BORON DILUTION

In these calculations version 3.02 of FLUENT has been used. FLUENT solves the equations for conservation of mass, momentum, and energy for the different cell volumes. Important for boron dilution is the diffusion model. In the presence of turbulence the turbulent diffusion model is used.

file ref. 40904-NUC 95-2738

In this model the diffusion flux J is given by:

$$J=\frac{\mu_t}{S_{ct}}\frac{\partial X}{\partial x}, \tag{4}$$

where μ_t is the turbulent viscosity, S_{ct} is the Schmidt number (default 0.7), X is the mass fraction of the chemical species considered, and x the spatial coordinate. The value of μ_t is calculated by the code and will depend on the turbulence model which is chosen. FLUENT has two turbulence models, the k-ε model, and the Algebraic Stress Model (ASM). The latter takes into account a directional dependence of the turbulent viscosity, but is less stable in complicated geometries. Because of this stability problem, the k-ε turbulence model has been chosen to model turbulence. In this model the kinetic energy k and the dissipation rate ε are calculated from transport equations for these variables, and then used to calculate the turbulent viscosity as:

$$\mu_t=\rho C_\mu\frac{k^2}{\varepsilon}, \tag{5}$$

where ρ is the density of the fluid, and C_μ is an empirical constant, which has the value 0.09. The turbulent viscosity is also used in the calculation of the Reynolds term in the momentum equation by applying the Bousinesque hypothesis (FLUENT, 1989).

The geometrical model in the FLUENT calculations comprises the inlet, the downcomer, the region under the core, the core itself, the region on top of the core, and the outlet. In total, 10164 cells are used in this model. The reactor core itself has been modelled as a porous medium, which results in a suppression of turbulence.

A problem in the FLUENT calculations is how the pump start-up should be modelled. It is impossible to model this completely due to limitations on CPU time. Instead, the dilution of boron is calculated quasi stationary. At discrete times the velocity field in the vessel is calculated, while the boron distribution in the vessel is kept constant. With this velocity field the boron dilution is calculated during a certain time interval, thus keeping the velocity field the same. The discrete times at which stationary velocity fields are calculated have been chosen in such a way that in every time interval 5 m^3 water is pumped into the reactor vessel.

To model the pressure drop in the vessel, the friction factor in the axial direction has been calculated in such a way that the frictional pressure drop is $3.04 \cdot 10^5$ Pa at hot full power (HFP). Because of geometrical considerations, the friction factors in the radial and tangential directions have been taken as 4 times the friction factor in the axial direction.

The flow through the reactor core has been modelled at an approximate value of 92 % of the total inlet flow. The remaining 8 % flows through the by-pass.

Furthermore, to allow modelling only half of the vessel, the angle between the inlet and outlet is taken to be 180 degrees. Although this is 90 degrees in reality, this does not significantly influence the results since the flow field in the core itself has almost no radial or tangential components.

Detailed results of the calculations can be found in reference (KEMA,1991). The results show a strong asymmetry in the problem, which is caused by the position of the inlet. Figure 2 shows the boron distributions after 9.6 seconds in the tangential planes in which the inlet and outlet pipes are situated, and in the radial plane at about 1/3 of the core. Figure 3 shows the same but now after 12.7 seconds. It can be seen in these figures that the inlet flow makes a big loop around the reactor core before it reaches the core bottom. Because mixing occurs along this rather long path, it is necessary to model this accurately. Also the side-flow under the core enhances mixing. In the core itself hardly any mixing takes place, because this is modelled as a porous medium in which turbulence is suppressed.

K-effective calculations with the different boron distributions in the reactor core, as they have been calculated by FLUENT, show that with a dilution of about 15 m^3 unborated water the reactor becomes critical (KEMA, 1992).

3 NUMERICAL DIFFUSION IN THE FLUENT CALCULATIONS

Scenario 3 has been reconsidered to asses the effect of numerical diffusion in the FLUENT calculations (KEMA, 1994a). Before presenting the results, it is worthwhile to pay some attention to the phenomenon of numerical diffusion, because it is a confusing subject. It is necessary to define in more detail what is meant by numerical diffusion in this context. In general, numerical diffusion is a severe inaccuracy, which has the same effect on the numerical solution as an artificial diffusion coefficient would have, provided a more appropriate numerical scheme would have been chosen.

Two different mechanisms for numerical diffusion can be distinguished. However, to illustrate the complexity of the matter, different authors give different subdivisions (Patankar, 1980; Leonard, 1979). First, in a multidimensional flow, numerical diffusion may occur when the calculational grid is not aligned with the direction of the flow. This effect is treated in more detail by Patankar, who refers to it as false diffusion. Leonard refers to it as streamline-to-grid-skewness. However, while Patankar mentions that a second numerical diffusion effect only exists in unsteady flows, Leonard states that the second effect is caused by a truncation error, and defines artificial diffusion coefficients both for time-independent and time-dependent flows by comparing the upstream differencing method with a central difference method. Patankar rejects this method, arguing that the central difference scheme cannot be used as a reference in a convection dominated problem. However, a model problem described by Leonard appears to show the correctness of the introduced artificial diffusion coefficient in a steady flow.

The boron dilution problem is a time-dependent problem. Apart from the numerical diffusion which might arise when the flow is not aligned with the grid, numerical diffusion might also arise due to other causes. Both Leonard (Leonard, 1979) and Patankar (Patankar, 1980)

file ref. 40904-NUC 95-2738

agree on this. Unfortunately Patankar does not give his view on this effect. Here, this effect is considered qualitatively. First, the Courant number C (Fletcher, 1991) is introduced:

$$C=\frac{u\Delta t}{\Delta x}, \tag{6}$$

in which u is the velocity of the flow, Δt the time step, and Δx the width of the mesh. This Courant number gives the ratio of the distance that information travels in one time step to the width of the mesh. In the case of a sharp transition of for example boron concentration it can be understood that the transport of this transition can only be described exactly with a suitable upwind scheme in the case that C=1. For Courant numbers smaller than 1 the transition will become less sharp. Numerical diffusion appears with the simple upwind scheme. We will call this effect artificial diffusion to distinguish it from false diffusion. In contrast with false diffusion, artificial diffusion may also appear in one-dimensional problems.

In this paper we consistently use the terms false and artificial when we want to indicate one of two different 'types' of numerical diffusion. In practice, it is difficult to separate them in time-dependent multi-dimensional problems such as the boron dilution problem, because then they may both appear.

By default, the power law interpolation scheme is used by FLUENT to determine the value of a variable on the boundary between two nodes. For diffusion dominated problems this scheme behaves as the central difference scheme, while for convection dominated problems the scheme behaves as the upwind scheme in which the value at a boundary is taken equal to the value upstream (Patankar, 1980). The power law scheme performs very well in one-dimensional steady problems. However, in two-dimensional problems numerical diffusion occurs when the calculational grid is not aligned with the direction of the flow. Because of the multidimensional nature of the flow, the power law scheme of FLUENT introduces false diffusion. A possibility for reducing false diffusion is the use of the so-called QUICK method (Leonard, 1979), in which a quadratic upwind interpolation scheme is used to calculate the convective flow. This method does not only reduce false diffusion, it also reduces the artificial diffusion.

Scenario 3 has been recalculated with FLUENT, using the QUICK numerical scheme. The results of the calculations are discussed in reference (KEMA, 1994). Here, the basic results of this work are presented.

In the calculations the spatial mesh has been doubled in all three directions, which also reduces numerical diffusion. This very fine mesh (81312 meshes) causes convergence problems in the region above the core. However, because this region is hardly of interest for the dilution in the core, it is modelled as a porous medium with outlet over the whole area at the top of the volume in the final FLUENT calculations.

Another difference with the calculations of section 2 is that now also flow in countercurrent direction through loop 2 is assumed (20 %). This means that only 72 % (instead of 92 %) of the inlet flow goes through the reactor core.

Figures 4 and 5 show the boron dilution in the core from the improved FLUENT calculations. It is clear that the boron dilution is more localized in the results with the fine mesh and the quadratic upwind scheme. This is just what is expected, since numerical diffusion has been reduced. LWRSIM calculations show that k-effective is about 4 % higher with the boron distribution from the improved FLUENT calculations, even though less inlet flow goes through the reactor core. The more localized boron dilution causes the reactor to become prompt critical for scenario 3. This shows that a proper calculation of the amount of mixing is very important in establishing the reactor dynamic response.

4 TRAC-PF1 CALCULATIONS ON BORON DILUTION

The results of the calculations performed by FLUENT indicate that boron dilution needs further attendance. Therefore, the same problem has been tackled with another code, TRAC-PF1 (LANL, 1992). TRAC stands for Transient Analysis Code, and is used to calculate transients in nuclear power plants. TRAC-PF1 is specifically dedicated to pressurized water reactors. The main advantages of TRAC-PF1 with regard to boron dilution problems are that it is suited to model the nuclear reactor system, and that it has the capability to calculate the reactor dynamic response, although this option has not been used in the calculations presented. The main disadvantages of TRAC-PF1 with regard to boron dilution problems are that it uses a coarse mesh, that it does not model physical diffusion processes, and that its Stability Enhancing Two-Step (SETS) numerical method (Mahaffy, 1982) is sensitive to artificial diffusion (Macian, 1995).

Scenario 3 has been recalculated with TRAC-PF1. For this calculation, the whole primary system of the reactor is modelled. An advantage of TRAC-PF1 is that it can simulate the start-up of the main circulation pump. Also, the counter-current flow through the other loop is calculated by TRAC-PF1. The ratio of the mass flow in loop 2 and the mass flow in loop 1 slowly increases until it reaches an equilibrium value of about 17 %, which is somewhat lower than the value assumed in the improved FLUENT calculations.

Figure 6 shows the nodalization of the vessel. Radially two nodes are used, one for the core, and one for the downcomer, while tangentially 4 equally spaced nodes are used. Axially the vessel is divided in 15 nodes.

Figure 7 shows the boron concentration in the bottom sectors of the core as a function of time as it is calculated with the improved FLUENT model, while figure 8 shows the same as it is calculated by TRAC-PF1. It is clear that the boron dilution is less localized in the TRAC-PF1 results, which results in lower k-effective values as calculated by LWRSIM. However, the nodalization in the TRAC-PF1 calculation is rather coarse. The main conclusion from this work is that this needs further investigation and that the TRAC-PF1 results are probably influenced by numerical diffusion.

file ref. 40904-NUC 95-2738

5 NODALIZATION STUDY WITH TRAC-PF1

As the study with TRAC-PF1 (section 4) clearly indicates, the nodalization used by TRAC needs further attendance. Therefore, the effect of different nodalization schemes on the boron distribution in the core has been studied. Again, this has been done for scenario 3. The detailed results of this study are given in reference (KEMA, 1994c). For this study three different nodalization schemes are used, the one given in figure 6, one scheme in which only two equally spaced angular nodes are used, and one scheme in which four radial nodes are used. The results indeed show that a finer nodalization scheme in the reactor vessel leads to a more localized boron dilution in the core, which results in a higher k-effective value. Especially the use of more radial zones leads to a more localized boron dilution. However, the TRAC-PF1 results seem to suffer from numerical diffusion, since the boron dilution is still less localized than in the FLUENT calculations, even though TRAC-PF1 does not model physical diffusion. Because of the velocity field in the core, artificial diffusion is especially expected in the axial direction. Unfortunately, the number of axial nodes has not been varied in this study.

6 CONCLUSIONS AND RECOMMENDATIONS

The general conclusions that can be drawn from the work done at KEMA can be summarized as follows:

1) From LWRSIM calculations with the FLUENT results, it can be seen that when the volume of unborated water in the primary loop is smaller than some limiting value between 10 and 15 m^3 water, the reactor with all rods in will not become critical. When LWRSIM calculations are done with the TRAC-PF1 results, this limit lies above 15 m^3, but this value is expected to be non-conservative due to numerical diffusion.

2) The FLUENT calculation with a fine mesh and the QUICK numerical scheme show that numerical diffusion may lead to a significant underestimation of k-effective values in the case of boron dilution.

3) Because of the fine mesh used in FLUENT, one may think of using a code as FLUENT for reference calculations. However, the FLUENT calculation does not take into account the actual start-up of the main circulation pump, and it does not calculate the mass flow through the downcomer and through the second loop. Furthermore, the detailed structures in the core have not been modelled.

4) The results with TRAC-PF1 seem to suffer from numerical diffusion problems in view of the large volumes of the meshes. This probably causes the main differences with the FLUENT results. Besides, TRAC-PF1 does not model physical (turbulent) diffusion.

For scoping analyses both FLUENT and TRAC-PF1 can be used to study boron dilution in a PWR.

file ref. 40904-NUC 95-2738

For more quantitative analyses an improved TRAC-PF1 model would be worthwhile at least in order to reduce numerical diffusion and possibly in order to model turbulent diffusion. Currently, such improvements are in course at PSU (Macian, 1995). Furthermore, it is recommended to set up a comparison between a system code such as TRAC-PF1 and a computational fluid dynamics code such as FLUENT. This comparison should be based on a problem that can be handled by FLUENT such as the boron dilution in a stationary velocity field. Care should also be taken to model the reactor vessel in more detail, which might be possible with the improved FLUENT code in which body fitted coordinates can be used. In this way it is not necessary to make additional assumptions in the FLUENT calculations. This may serve as a verification of an improved TRAC-PF1 model so that afterwards scenarios in which for example the start-up of the pump and the dynamic response of the reactor are taken into account, can be tackled with confidence by TRAC-PF1. To model the reactor dynamic response the 3-dimensional neutronics capability of TRAC-PF1/NEM (Bandini, 1990) will be very useful.

A major issue that remains is how to establish realistic scenarios for boron dilution in a PWR. We have merely made assumptions regarding a volume of unborated water hypothetically present in one loop of the primary system with a particular type of scenario. In fact, the choice of scenarios will determine what degree of sophistication for the calculational method is appropriate.

Acknowledgement

This work has been funded by the electricity production sector of the Netherlands.

REFERENCES

Bandini B. (1990), A Three Dimensional Transient Neutronics Routine for the TRAC-PF1 Reactor Thermal Hydraulics Computer Code, PhD Thesis, The Pennsylvania State University, USA.

Fletcher C.A.J. (1991), Computational Techniques for Fluid Dynamics, Volume I, Springer-Verlag, Germany

FLUENT (1989), "Computer Programs for Simulating Fluid Flow", creare.x Inc., Hanover NH, USA

KEMA, 1991 (K.P. Termaat). Boorzuurverdunning bij PWR's; berekeningen met het modelleringspakket FLUENT V3.02, report 60118-TFO 91-3036.

file ref. 40904-NUC 95-2738

KEMA, 1992 (W.J. Oosterkamp, K.P. Termaat, F.C.M. Verhagen). k-ε modeling of deboration transients in a PWR, handout at ANS/ENS Int. Mtg. November 1992.

KEMA, 1994a (K.P. Termaat and F.C.M. Verhagen). Modelleringsinvloeden bij stromings- en reactiviteitsberekeningen van boriumtransiënten in PWR's, report 40339-NUC 93-7320.

KEMA, 1994b (J. Wesseling). Analysis of a deboration transient in a PWR using TRAC/PF1, report 40380-NUC 93-7314.

KEMA, 1994c (D. Knepper). Vessel nodalization study of a deboration transient in a PWR using TRAC/PF1, report 40900-NUC 94-2787.

Leonard B.P. (1979), A stable and accurate convective modelling procedure based on quadratic upstream interpolation. In: comp. meth. in appl. mech. and eng. 19, p. 59

Los Alamos National Laboratory (1992), TRAC-PF1/MOD2 manuals, Vol. I-IV

Macian R. (1995), Improvements to the One Dimensional Solute Tracking Algorithm of TRAC-PF1, TRAC-BF1 User's Group Meeting 1995, Penn State University, State College PA, USA.

Mahaffy J.H. (1982), A Stability Enhancing Two-Step Method for Fluid Flow Calculations. In: J. comput. phys. 46, p. 329

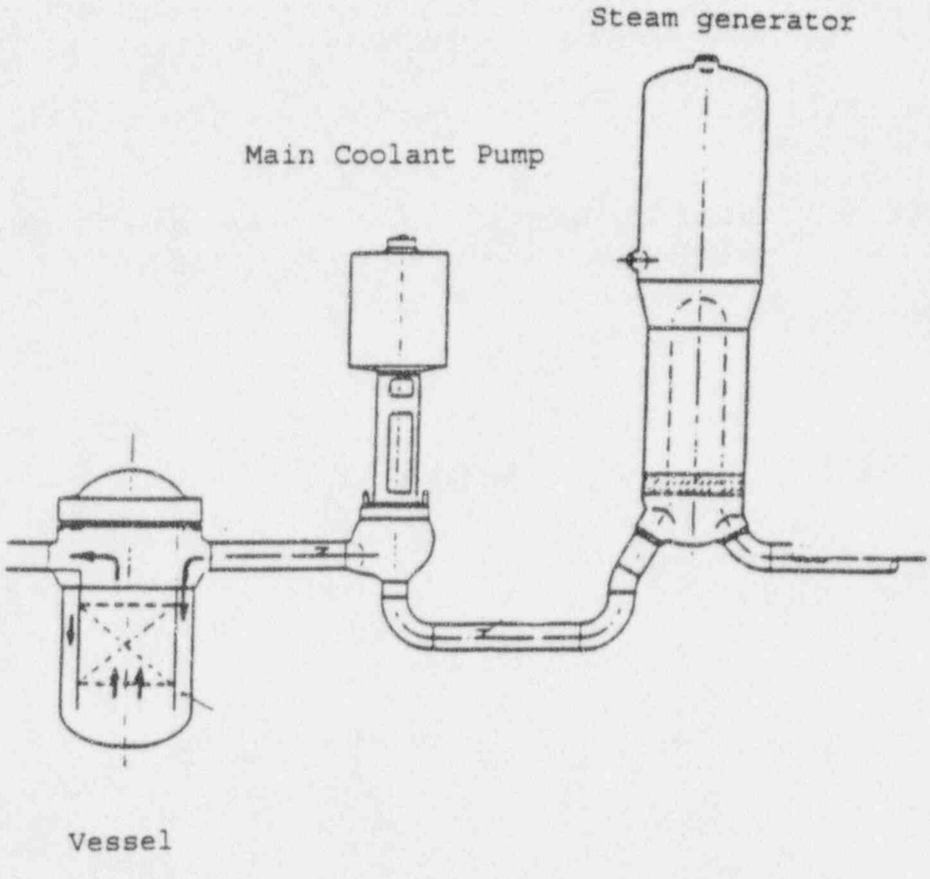
Patankar S.V. (1980), Numerical Heat Transfer and Fluid Dynamics, Series in Computational Methods in Mechanics and Thermal Sciences, Hemisphere Publishing Corporation, USA.

file ref. 40904-NUC 95-2738

Table 1: Overview of calculations.

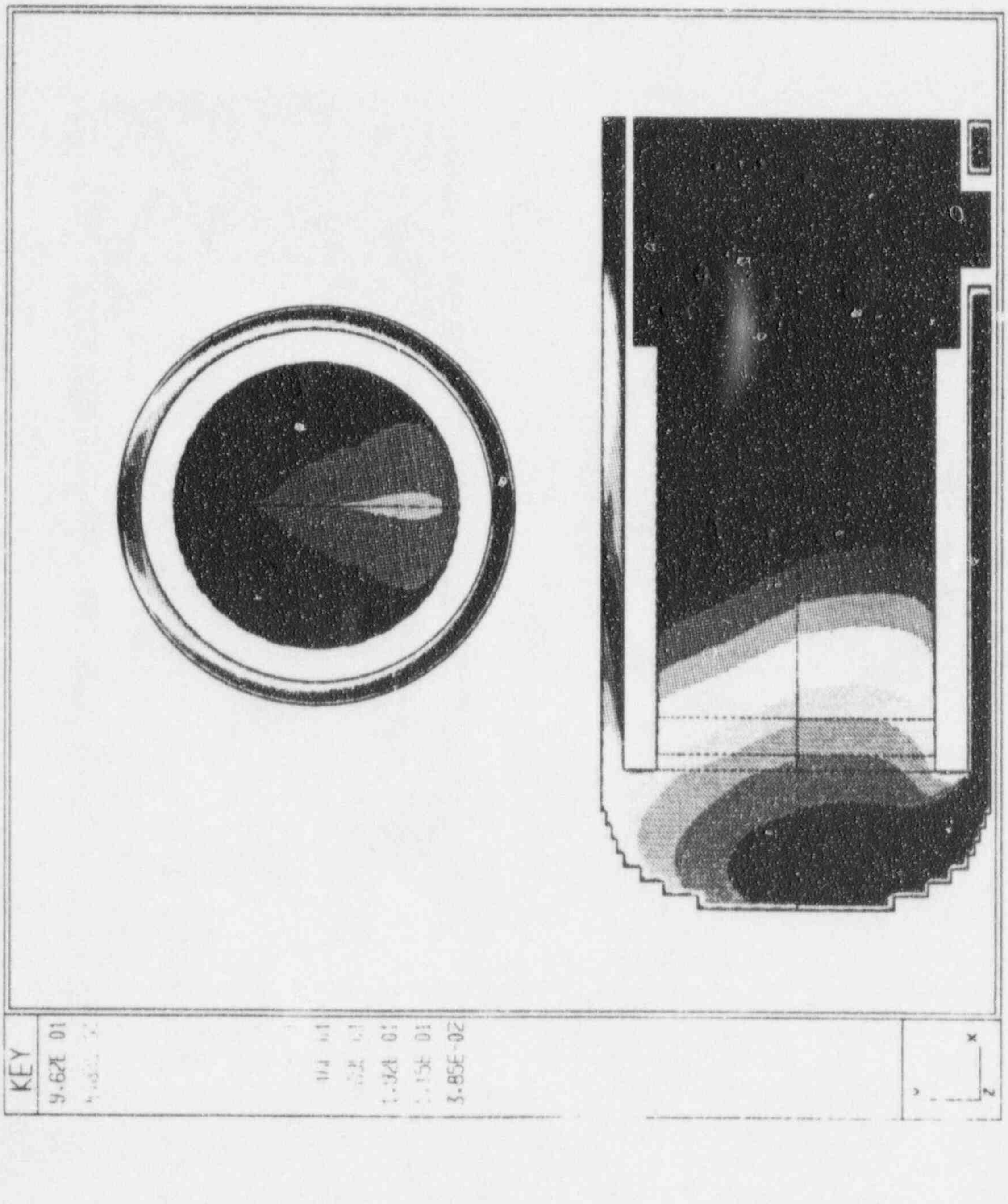
Section	Code	Scenarios	reference	number cells
2	FLUENT 3.02	0,1,2,3	KEMA 1991	10164
3	FLUENT 3.02	3	KEMA 1994a	81312
4	TRAC-PF1	3	KEMA 1994b	120 (vessel)
5	TRAC-PF1	3	KEMA 1994c	60,120,240

Figure 1: Schematic drawing of one loop in the primary system.



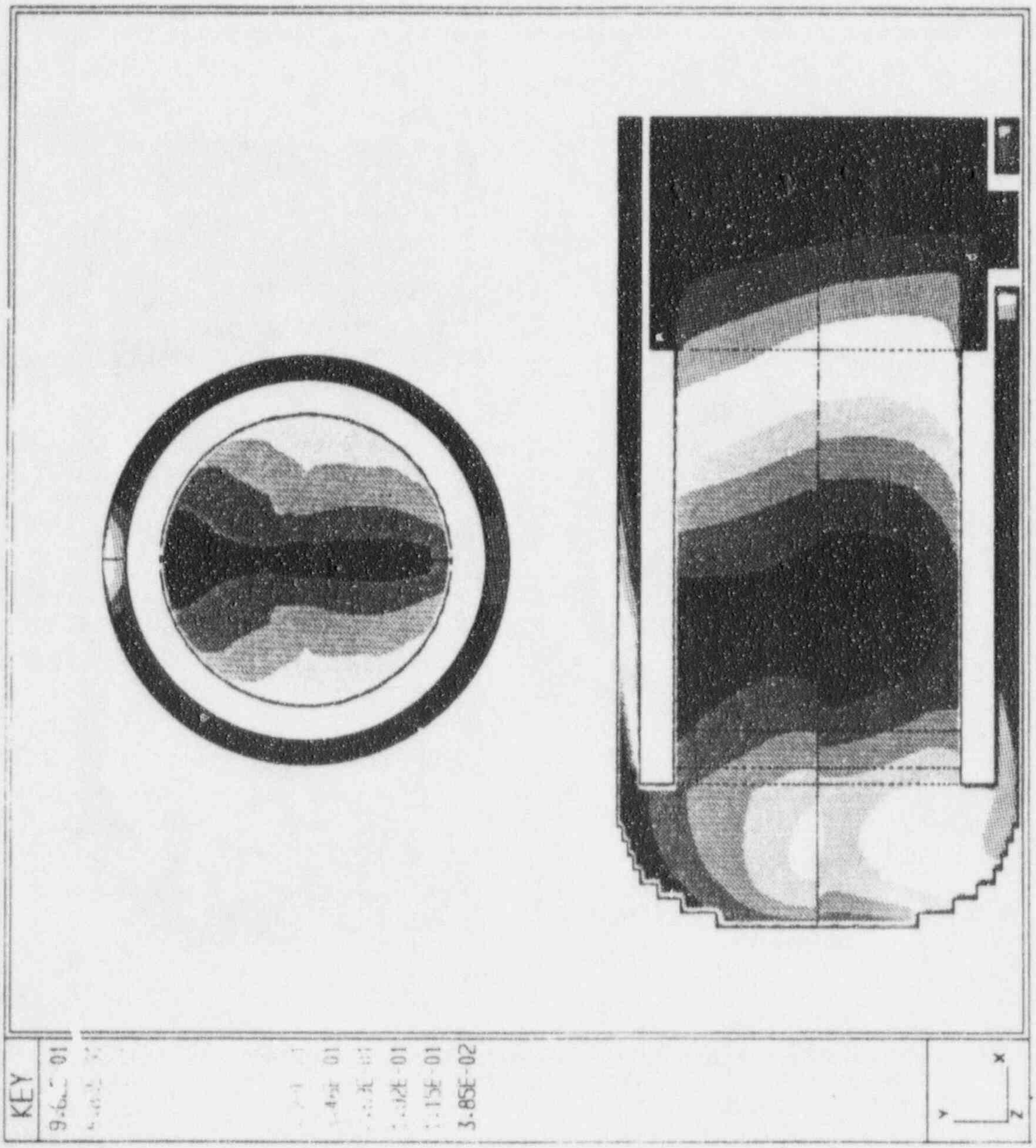
file ref. 40904-NUC 95-2738

Figure 2: Relative boron concentrations after 9.6 seconds for scenario 3 in the tangential planes of in- and outlet, and in the radial plane situated at about 1/3 of the core; FLUENT results from reference (KEMA, 1991).



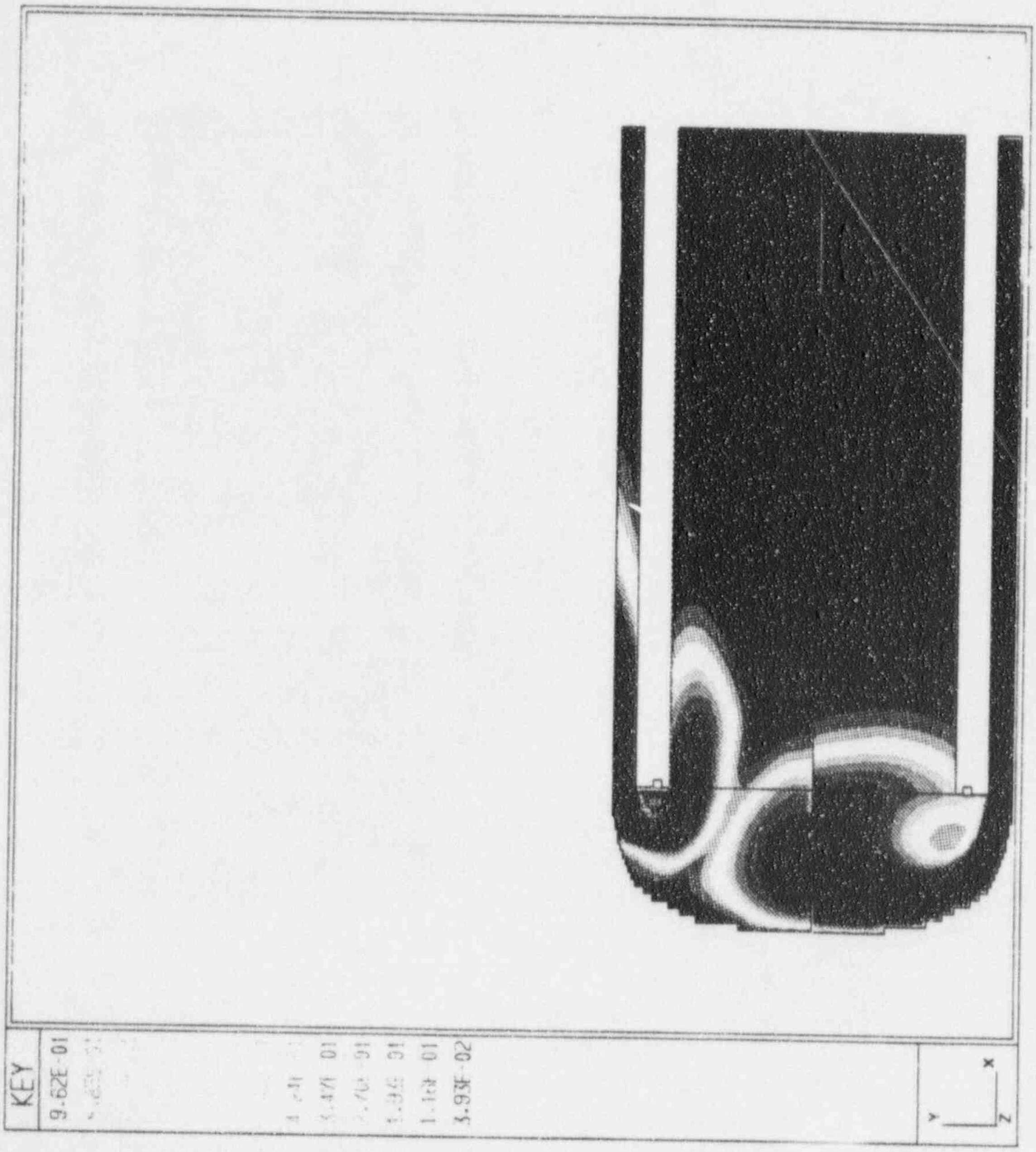
file ref. 40904-NUC 95-2738

Figure 3: Relative boron concentrations after 12.7 seconds for scenario 3 in the tangential planes of in- and outlet; and in the radial plane situated at about 1/3 of the core; FLUENT results from reference (KEMA, 1991).



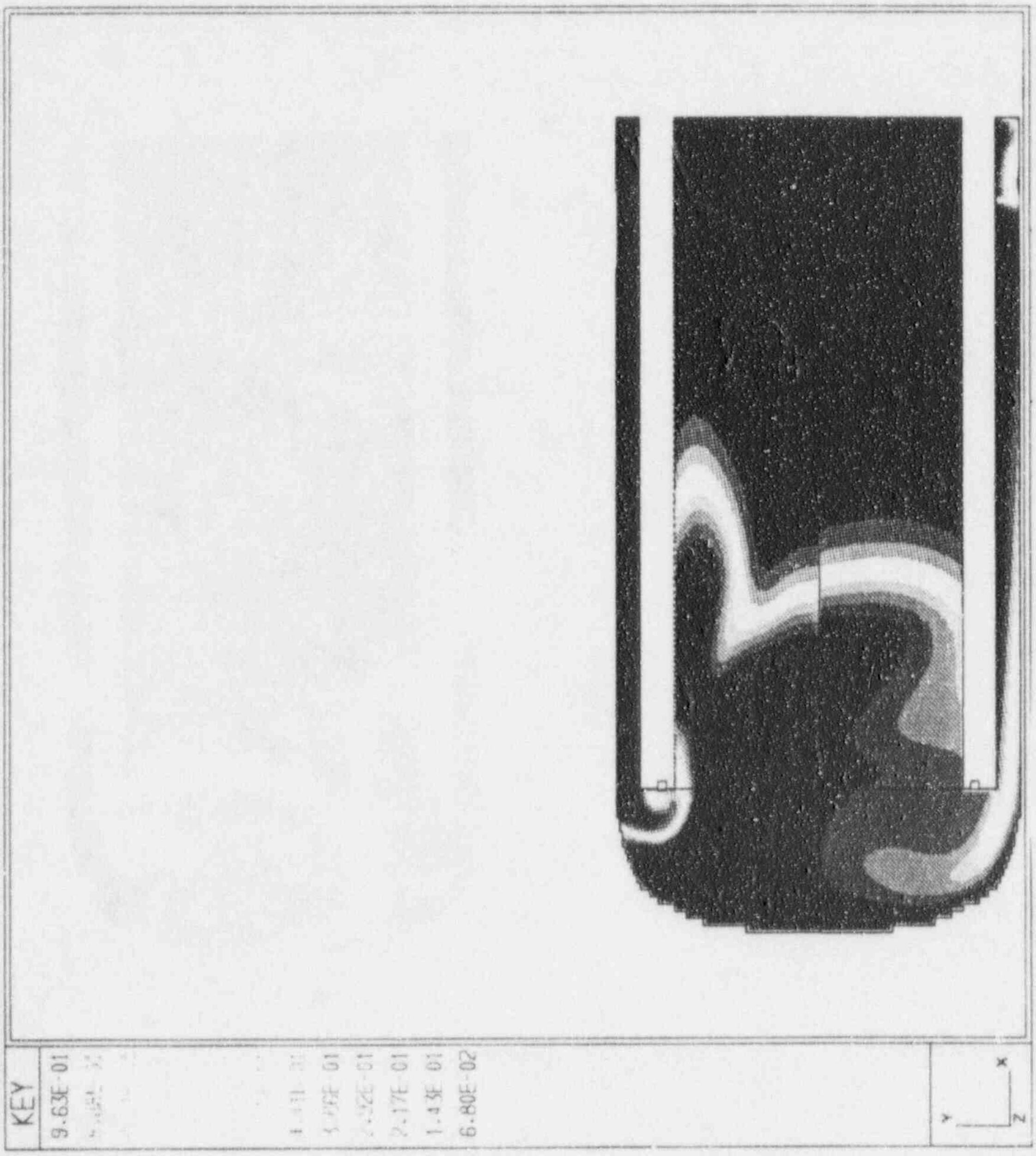
file ref. 40904-NUC 95-2738

Figure 4: Relative boron concentrations after 9.6 seconds for scenario 3 in the tangential planes of in- and outlet; improved FLUENT results from reference (KEMA, 1994a).



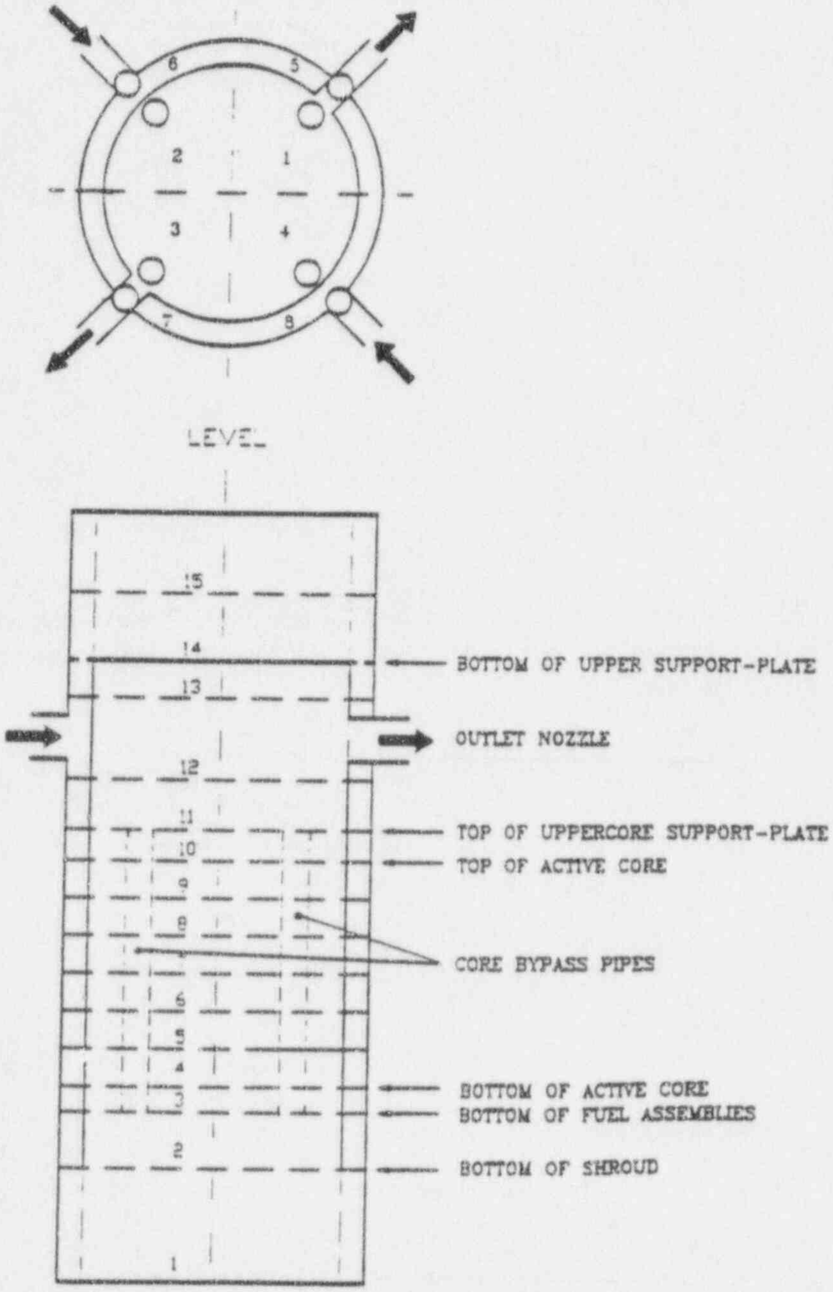
file ref. 40904-NUC 95-2738

Figure 5: Relative boron concentrations after 12.6 seconds for scenario 3 in the tangential planes of in- and outlet; improved FLUENT results from reference (KEMA, 1994a).



file ref. 40904-NUC 95-2738

Figure 6: Nodalization scheme for the vessel in the TRAC-PF1 calculations of reference (KEMA, 1994b).



file ref. 40904-NUC 95-2738

Figure 7: Boron concentrations as a function of time for scenario 3 in the region under the core; improved FLUENT results from reference (KEMA, 1994a). Because of symmetry the curves for sectors 1 and 3 are identical.

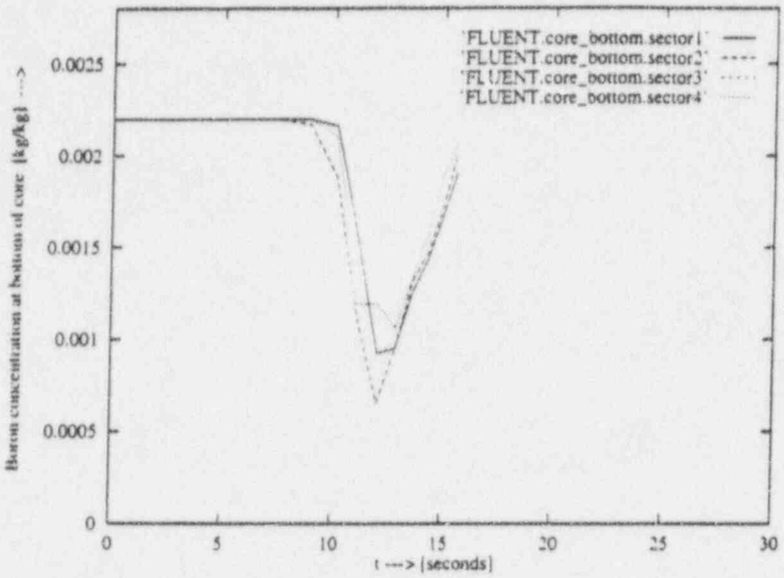
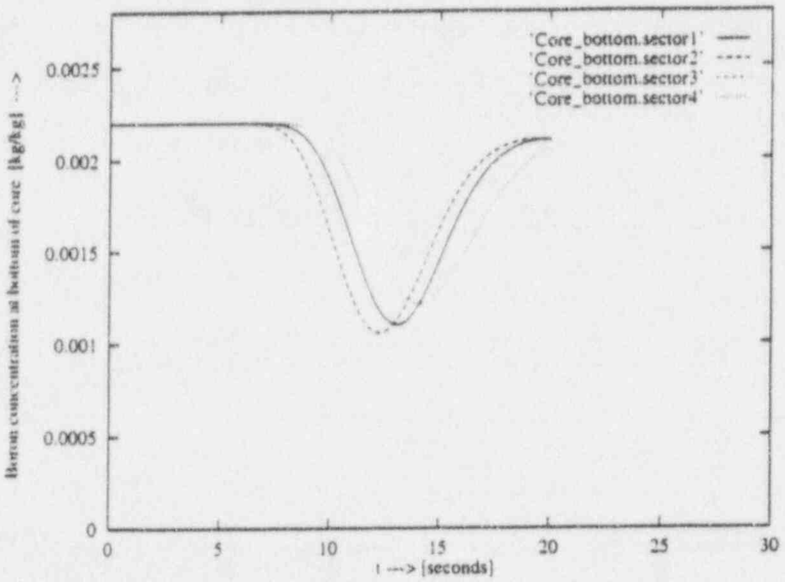


Figure 8: Boron concentrations as a function of time for scenario 3 in the region under the core; TRAC-PF1 results from reference (KEMA, 1994b). Because of symmetry the curves for sectors 1 and 3 are identical.



file ref. 40904-NUC 95-2738

APPLICATION OF NUMERICAL MODELLING FOR STUDYING BORON MIXING IN LOVIISA NPP

presented at

OECD Specialist Meeting on
Boron Dilution Reactivity Transients

State College, PA USA
18th-20th October 1995

Peter Gango

IVO INTERNATIONAL LTD
Rajatorpantie 8, Vantaa
FIN-01019 IVO, Finland

tel. +358-0-8561 2352
fax +358-0-8561 3403

APPLICATION OF NUMERICAL MODELLING FOR STUDYING BORON MIXING IN LOVIISA NPP

Peter Gango
IVO International Ltd
01019 IVO, Finland

ABSTRACT

A program is going on for studying numerically boron mixing in the downcomer of Loviisa NPP (VVER-440). The commercial general purpose CFD code PHOENICS is used for solving the governing fluid flow equations in the downcomer geometry of VVER-440. So far numerical analyses have been performed for pump driven forced convection transients and some natural circulation scenarios. In this paper the validation of the buoyancy extended numerical model against PTS mixing experiments is reported. Additionally the results of one natural circulation case for the real reactor are given.

1 INTRODUCTION

Inhomogenous boron dilution has recently become one of the most important issues in PWR safety. It has been found out that certain operational or accidental conditions might induce formation of slugs with low boron concentration in the primary circuit loops of a PWR [1,2,3]. If such a slug is transported unmixed to the core there is a risk for a reactivity accident. This transport may occur during a Reactor Coolant Pump (RCP) start or reestablishment of natural circulation. However, in many scenarios the boron content of the diluted slug is supposed to increase during the transport due to turbulent convective mixing in the downcomer. Thus mixing serves as an inherent protection mechanism, which diminishes the risk and might eliminate it completely.

At present there are two possible methods available for studying mixing in the downcomer of a PWR; scaled down model experiments and numerical simulation. The scaling principles for forced flow conditions allow relatively small scale models to be used (1/5), but in natural circulation conditions large and expensive test facilities are needed for a proper scaling. The rapid development of computers has made it possible to make use of Computational Fluid Dynamics (CFD) in studying large 3D problems. However, in nuclear applications the numerical models require experimental validation to be accepted as tools for solving crucial problems, even if commercial codes were used. Therefore at present the most common procedure for studying mixing of boron in a PWR downcomer is a combined experimental and numerical approach [3,4]. Small scale model experiments are used for code validation purposes and in the cases were scaling problems exist, the scaling is done by the numerical model. However, the design of some of the current experimental facilities does not enable producing any results fully transferable to the real reactor. Thus the results from those experiments serve <u>only</u> as validation data for numerical models [5].

Loviisa NPP is owned and operated by Imatran Voima Oy (IVO) and consists of two Russian VVER-440 type reactor units, 445 MWe each. The primary circuit lay-out is shown in figure 1 and the pressure vessel and its internals in figure 2. The total coolant volume of the primary circuit is about 200 m^3 (pressurizer not included). The six primary loops have a volume of 16 m^3 each. The volumes of the downcomer and the lower plenum are 20 m^3 and

27 m^3 respectively. The large coolant volumes in the downcomer and in the lower plenum compared to the loop volume (ratio 3:1) show that a considerable mixing potential exists. This favorable geometrical feature is specific to the VVER-440 design; a typical volume ratio for western PWRs is 1:1.

This paper describes the current approach applied at Loviisa NPP for studying boron mixing in the downcomer of the reactor. So far numerical analyses have been performed for two different pump driven forced convection scenarios and a few buoyant natural circulation cases. Additionally, steady state analyses with full flows in all loops were performed and compared to existing experimental data during the initial development of the model [6]. Some results from the different pump driven cases are reported earlier [5]. The current paper focuses on validation of the buoyancy extended numerical model against mixing experiments performed for studying Pressurized Thermal Shocks (PTS) caused by cold High Pressure emergency coolant Injection (HPI). A comprehensive series of such experiments for VVER-440 geometry was performed at IVO in the mid 80s [7]. In addition to the PTS simulation results one natural circulation case for the real reactor is reported.

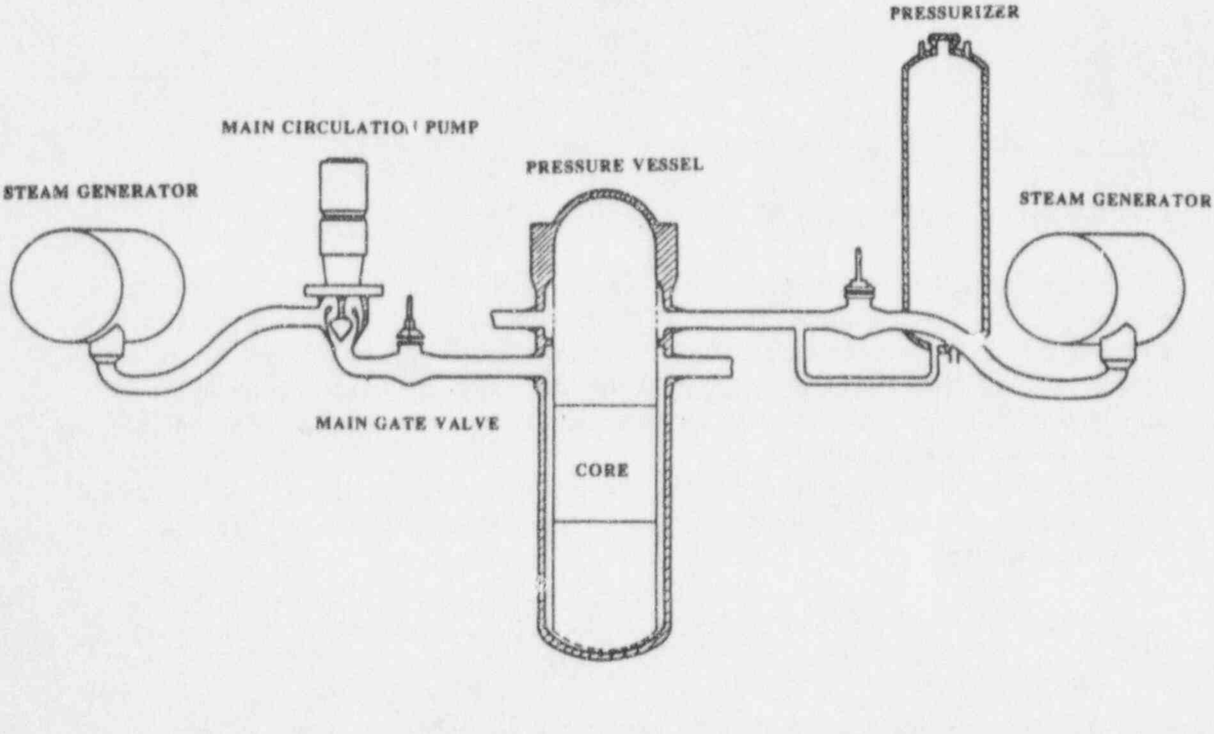


Figure 1. Loviisa 1&2, primary circuit lay-out

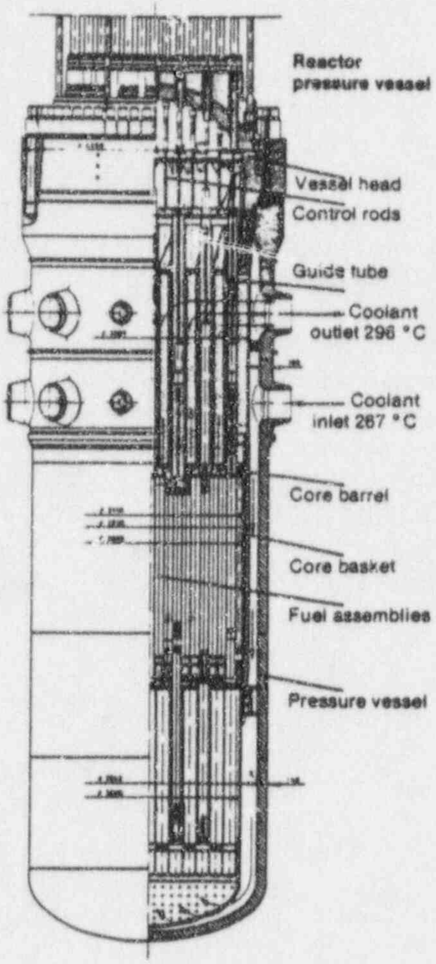


Figure 2. Loviisa 1&2, pressure vessel and internals

2 NUMERICAL APPROACH

In our numerical modeling approach the PHOENICS code is used for solving the governing fluid flow equations. In the developed model the time dependent, Reynolds averaged elliptic Navier-Stokes equations for uncompressible flow are solved by using implicit time integration. The body-fitted co-ordinate option in the code is made use of to reproduce the complex geometry of the downcomer [8]. The number of cells in the grids is typically between 50000 - 100000.

The mixing of boron in the downcomer occurs by convection, that is by the combined process of advection and diffusion. The general conservation equation for a scalar C in a flow field can be written as

$$\frac{\partial \rho C}{\partial t} + u_i \frac{\partial \rho C}{\partial x_i} = \frac{\partial}{\partial x_i}\left(D \frac{\partial \rho C}{\partial x_i}\right) + S \, , \tag{1}$$

where D is the diffusion coefficient, S the source term and subscript i refers to cartesian co-ordinate direction. Although the advection and diffusion terms are treated separately in the equation the mixing process itself is a complex combination of the phenomena. In turbulent flows the local concentration gradients are smoothened out by turbulence and finally by molecular diffusion. In typical downcomer flow conditions (operation with pumps or natural circulation) the molecular effects are negligible from the large scale mixing point of view.

In modeling of the mixing process most commonly the effect of turbulence on mixing is implemented in the diffusion coefficient. This effective diffusion coefficient is highly dependent on flow conditions. In CFD usually the local diffusion coefficient is calculated with a given turbulent Schmidt number from an eddy viscosity field produced by the turbulence model used. In more advanced turbulence models the eddy viscosity concept and the turbulent diffusion coefficient are not used, instead direction dependent turbulent fluxes are calculated [9].

The "traditional" first order upwind discretisation method for the advection terms in the equations is known to produce artificial smoothing or numerical diffusion in coarse grid solutions. The role of numerical diffusion in boron mixing can be controlled by densing the calculational grid, but the current computer capacity sets limits to the usage of this procedure. To diminish the undesired effect of numerical mixing, built-in higher order discretisation scheme options in PHOENICS were taken in use. Second or third order discretisation was used for all momentum equations and scalar equations including the turbulence quantities. The usage of higher order discretisation schemes, however, does not completely solve the problem of numerical diffusion. The role of numerics in the solution must be found by grid independency studies and time step variation. The numerical effects have to be small or insignificant to prevent false conclusions about the efficiency of mixing.

At this stage only the built-in standard k-ε model has been used for turbulence modeling. The k-ε model is known to have weaknesses, but the simple structure compared to more advanced models makes its usage attractive. The most severe uncertainties concerning the usage of k-ε turbulence model in the mixing studies come from the behavior in rapidly accelerating flows and from the well known difficulties in modeling buoyancy induced turbulence.

3 SIMULATION OF IVO PTS MIXING EXPERIMENTS

In 1983 a comprehensive series of experiments was performed for studying cold HPI water mixing in the cold leg and downcomer of the reactor in connection to the PTS problem [7,10]. The experiments were performed at IVO Hydraulic Laboratory in a 2/5 scale semiannular test facility with a geometry similar to Loviisa reactors (VVER-440). These experiments were found to be suitable for numerical simulation and serve as experimental reference for a validation of our buoyancy extended numerical model.

The numerical simulation series consisted of totally 9 different calculations for three chosen PTS mixing experiments. In the series special effort was put on studying the effect of numerics; different turbulence models were not tested at this stage. The influence of chosen simulation parameters is here only summarized and conlusions are drawn concerning model applicability for boron mixing studies in natural circulation conditions.

3.1 Description of the modelled event

In PWR PTS -scenarios thermal shocks caused by cold HPI coolant are assumed to threaten the integrity of the reactor pressure vessel (RPV). During normal operation the coolant temperature in the downcomer is about 260 °C (≈ RPV wall temperature). At Loviisa NPP the temperature of the emergency coolant tank is 40 - 60 °C. Unmixed transport of cold HPI coolant into brittle areas of the hot pressure vessel might be hazardous. However, it has been experimentally shown that efficient mixing of the cold HPI water occurs. Thus mixing serves as an inherent safety feature against PTS.

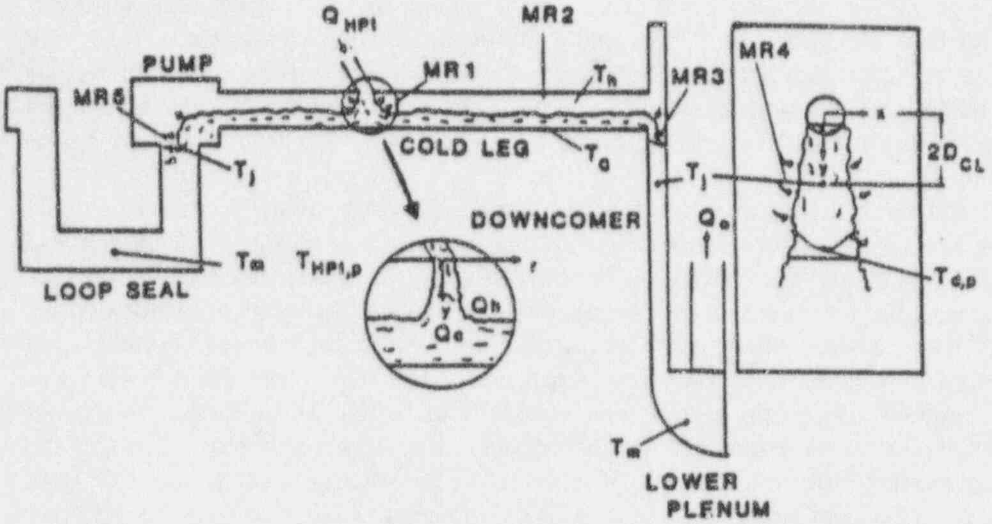


Figure 3. Mixing regions of HPI coolant in PTS conditions /11/

The mixing of cold HPI coolant injected into the primary circuit occurs in several regions (figure 3). The cold leg area close to the HPI inlet nozzle comprises the first mixing region (MR1). The mixing rate in MR1 depends on flow conditions and nozzle geometry. With high HPI flow rates (compared to loop flow) the cold leg area close to the nozzle is completely filled with HPI water and no mixing occurs in MR1. In the cold leg between the HPI nozzle and the downcomer the flow is stably stratified and mixing between the layers is limited. Accordingly the mixing occuring in region MR2 can be neglected. In the downcomer entrance region mixing occurs due to the rapid acceleration of the flow (MR3). The cold HPI plume decay in the downcomer constitutes mixing region MR4.

3.2 Modelling aspects

At IVO a regional mixing model (REMIX) is used for calculating the mixing of HPI coolant for PTS applications. Numerical 2D and 3D simulations have not been performed at IVO, but others have applied the numerical approach [12,13].

The increased speed of computers and development of CFD codes have made it possible to revise the numerical simulation of PTS mixing. Today it is possible to solve the governing flow equations in body-fitted grids eliminating errors arising from an inaccurate description of the geometry. Additionally more accurate numerical schemes are available to diminish the effect of numerics in the solution.

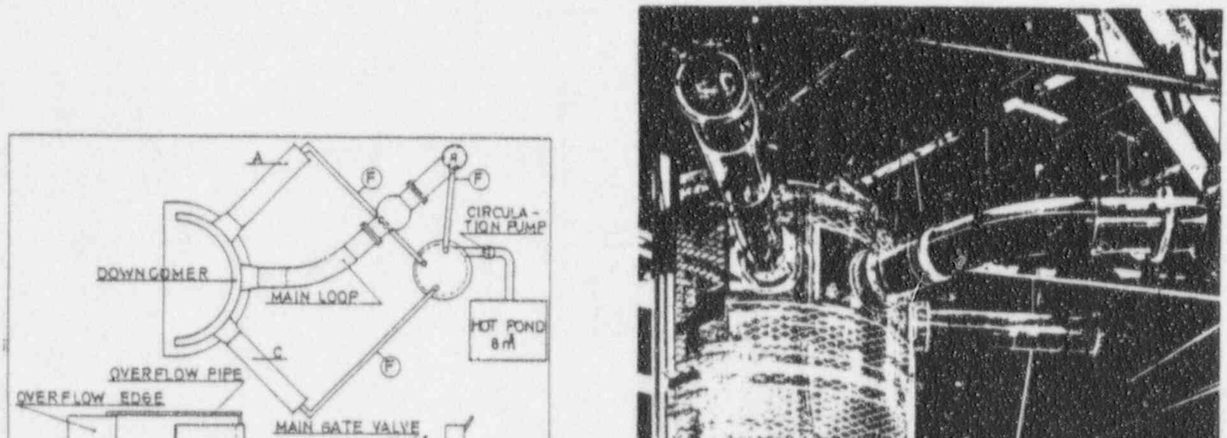
However, even today the lack of computer resources enforce us to solve the equations in coarse grids compared to the smallest details of the flow field. The flow equations have to be solved time-averaged and the effect of turbulence is taken into account by a turbulence model implemented in the code. In fact, today the largest source of error concerning simulation of buoyant flows lies in the modeling of turbulence. Models which assume an isotropic structure of turbulence, such as the standard k-ε model, are known to perform badly in buoyant flow conditions, were the structure of turbulence is strongly unisotropic. Furthermore the Reynolds analogy between momentum and scalar transport is known to fail in strongly buoyant conditions.

3.3 Experimental work

The test rig of IVO PTS mixing experiments is shown in figure 4a. The 2/5 scale semiannular downcomer and the cold legs were made of transparent material to enable visual observations (figure 4b). The HPI nozzle was located in the second of the three cold legs.

In the experiments the temperature of the HPI water was $T_{HPI} = 10$ °C. The hot primary coolant was modelled with water of temperature $T_{PC} = 80$ °C. In several experiments salt was added to the HPI water to increase the density difference. The flow conditions and the density difference were varied. Different flow conditions were achieved by changing the flow rates of the loops and the HPI. The test matrix of the experiments is shown in table 1.

For the simulations test 33 was chosen as reference test without salt and test 22 as reference with salt. The densimetric Froude numbers for the HPI in the experiments were $Fr_{CL,\ HPI} = 0.646$ and $Fr_{CL,\ HPI} = 0.253$ respectively. In addition test 47 was chosen as reference test without both side loop flows and salt (Fr same as in 33).



Figures 4a and 4b. Schematic view and photo of the IVO PTS mixing facility

Table 1. Test matrix of IVO PTS mixing experiments /7/

Test nr	Q_{HPI} 1/s	$Q_{L,A}$ 1/s	$Q_{L,B}$ 1/s	$Q_{L,C}$ 1/s	$Fr_{CL,HPI}$	Salinity $\Delta\rho/\rho$
3	2.31	0	1.87	0	0.379	0
4	2.31	1.87	1.87	1.87	0.376	0
7	2.02	1.87	1.87	1.87	0.129	1
8	2.00	0	1.87	0	0.129	1
9	2.02	0	0	0	0.130	1
10	2.31	0	0	0	0.147	1
12	0.62	0	0	0	0.040	1
13	0.62	0.62	0.62	0.63	0.040	1
14	0.62	0	0.62	0	0.039	1
15	0.62	1.87	1.87	1.87	0.040	1
16	0.31	0	0	0	0.020	1
19	0.10	0.3	0	0.3	0.006	1
20	2.31	1.87	0	1.87	0.146	1
21	2.31	1.87	1.87	1.87	0.147	1
× 22	4.0	2.0	0	2.0	0.253	1
23	0.20	0.3	0	1.0	0.013	1
26	0.62	1.0	0	1.5	0.096	0
27	0.62	0.62	0.62	0.62	0.101	0
28	0.20	0.3	0	1.0	0.032	0
30	1.25	1.87	0	1.87	0.202	0
31	0.62	1.87	0	1.87	0.100	0
32	0.10	0.3	0	0.3	0.016	0
× 33	4.0	2.0	0	2.0	0.646	0
34	1.25	1.87	0	1.87	0.126	1/4
35	2.31	1.87	0	1.87	0.188	1/2
36	0.62	1.87	0	1.87	0.050	1/2
38	1.25	1.87	0	1.87	0.102	1/2
40	0.20	0.3	0	1.0	0.016	1/2
41	1.25	1.87	0	1.87	0.088	3/4
42	1.25	1.87	0	1.87	0.080	1
43	4.0	2.0	0	2.0	0.323	1/2
44	4.0	0	0	0	0.324	1/2
45	4.0	0	0	0	0.255	1
46	3.0	2.0	0	2.0	0.477	0
× 47	4.0	0	0	0	0.644	0
48	2.0	1.87	0	1.87	0.318	0
49	2.31	1.87	0	1.87	0.366	0
50	0.62	0	0.62	0	0.100	0
51	2.31	0	0	0	0.372	0
52	0.62	0	1.87	0	0.100	0

Salinity
0 : $\Delta\rho/\rho = 0.02$
1/4: $\Delta\rho/\rho = 0.06$
1/2: $\Delta\rho/\rho = 0.10$
3/4: $\Delta\rho/\rho = 0.13$
1 : $\Delta\rho/\rho = 0.16$

3.4 Numerical model

The body-fitted grid option in PHOENICS was made use of to reproduce the geometry of the downcomer in the facility. The length of cold leg piping included was one to two cold leg diameters. The grid density in different coordinate directions was varied to find out possible grid density dependences; the total number of cells used in the grid was between 16128 and 29744. Symmetry was made use of in the model by setting the symmetry axis in the middle of the second loop. One of the grids used in the simulation series is shown in figure 5.

The inflow boundaries of the model were set according to the experiment to be simulated. At inlet the turbulence quantities were varied corresponding to turbulence intensity between 4 and 10 %. The outlet of the model was set inside the perforated plate of the vessel bottom by fixing the pressure. The perforated plate was modelled with porosity factors and a quadratic friction pressure loss source term. The standard logarithmic wall function approach was applied at the adiabatic walls of the model. At the symmetry boundary PHOENICS automatically sets zero flux conditions for all solved variables.

In the numerical model PHOENICS was put to solve the time dependent time-averaged elliptic fluid flow equations for mass continuity, momentum in 3d, turbulence quantities k and ε, and the scalars for energy and salt. Additionally the variables temperature, density and eddy viscosity were calculated and stored.

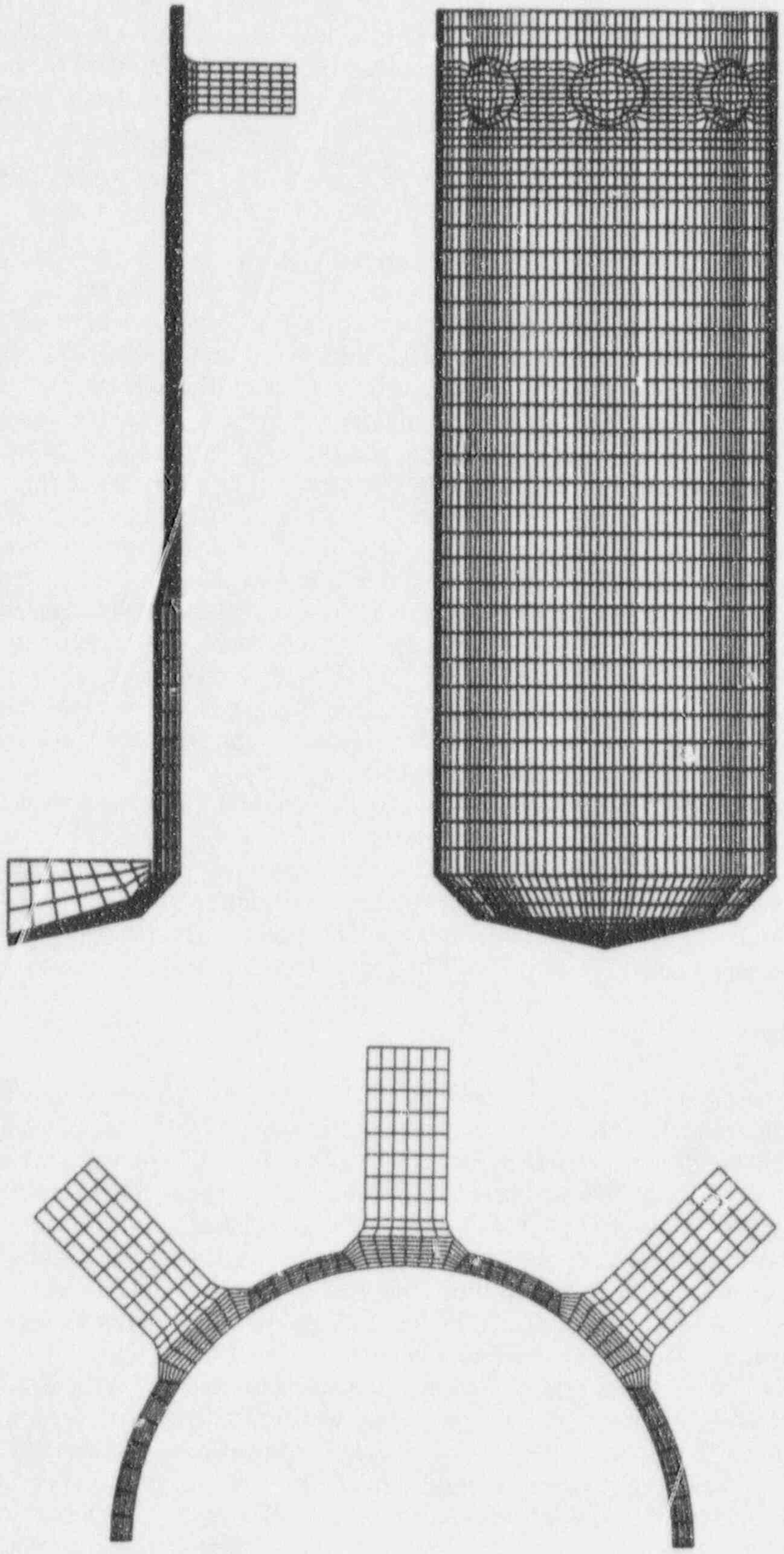


Figure 5. Computational grid for PTS simulations

To take into account buoyancy in the model two separate measures were taken. Firstly, the gravity force was introduced in the momentum equations by activating the built-in source term option in PHOENICS. Cell density was calculated from cell temperature T and salt concentration C_s with the approximative formula

$$\rho = 1001 - 0.003645 \cdot T^2 - 0.07018 \cdot T + 333.3 \cdot C_s^2 + 836.7 \cdot C_s$$

Secondly, the so called buoyancy terms were activated in the equations of the turbulence quantities. The purpose of the buoyancy terms is to dampen out turbulence in stably stratified flows and to increase turbulence in unstable stratification conditions. The additional terms increase the performance of the standard k-ε model in buoyant flow conditions.

In the simulation series first and second order discretisation schemes for the convection terms were compared to find out the effect of numerics in the solution. The sensitivity to time step length was studied by varying between $\Delta t = 0.1$ s and $\Delta t = 0.5$ s. In addition the effect of turbulent Prandtl number was studied by giving it two different values ($Pr_t = 0.5$ and $Pr_t = 0.8$).

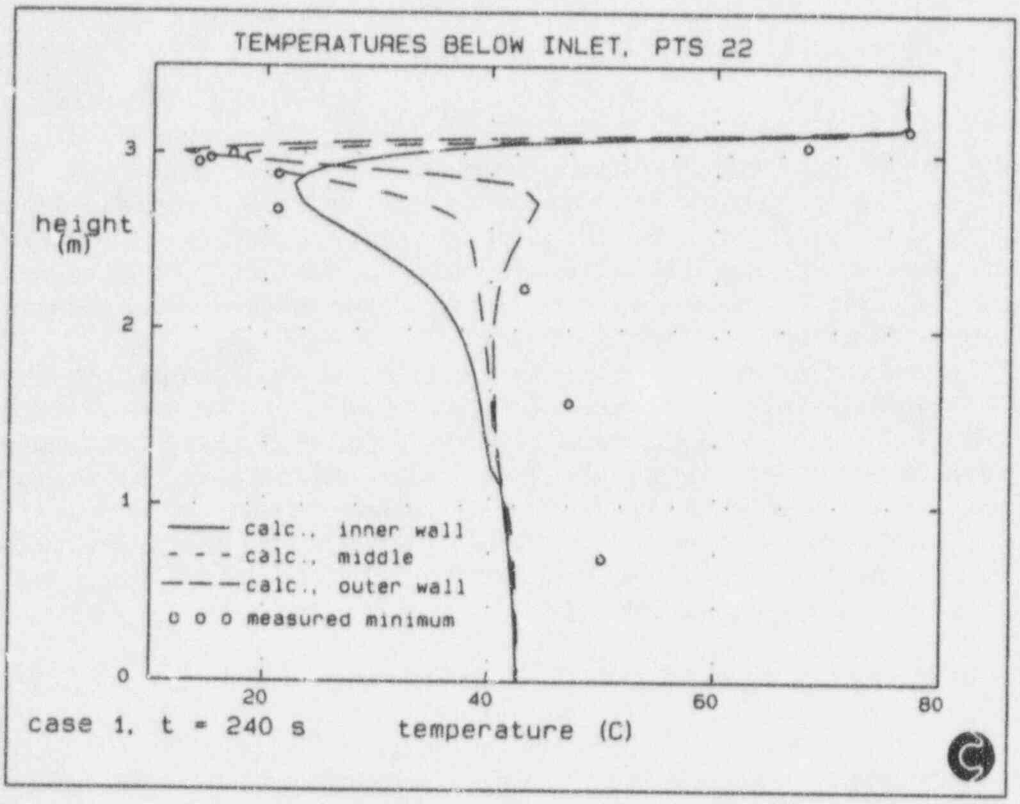
In the cases of experiments 22 and 33 the simulation was started with a steady state calculation for the initial flow conditions, i.e side loop flows equal to 2 dm^3/s and a zero flow in the HPI loop. The simulation of experiment 47 started from completely stagnant conditions. The transient part of the simulation was started by introducing hot water inflow in the HPI loop. The hot water injection stood for the original content of the cold leg. The temperature history used at model inlet was only approximative, because mixing in the HPI nozzle region at the very beginning of the transient was not included in the numerical model. The HPI rate was equal to 4 dm^3/s in all three experiments.

The total simulation time was varied in the different cases, the longest simulations lasted 4 minutes from the start of HPI. In most cases only the first 45 to 60 seconds of the transient were simulated. The differences in the simulations became apparent during this initial part of the transient when the cold HPI front travelled from the cold leg to the bottom of the vessel. In the simulation of experiment 47 the calculation was stopped at 60 s, because the experimental results showed strong asymmetry after 1 minute.

3.5 Results

Some resulting flow and temperature fields from the different simulation cases are shown in appendix 1. The calculated temperatures below HPI loop were compared to the measured minimum temperatures at each level; results from two of the cases are shown in figures 6a and 6b. In the results the cold HPI front can be seen to move towards the bottom of the vessel quite fastly. The flow pattern with strong swirling at the sides of the front reminds of the mushroom shaped cloud from a strong explosion. Only limited differences could be seen in the flow pattern of simulation PTS 47 compared to the cases with side loop flows. The small difference is caused by the fact, that without side loop flows all the fluid that is mixed into the plume comes from below.

The front behavior is case dependent and none of the simulation cases showed exactly the same behavior as was seen in the photos and videos from the experiments. In the experiments the plume decay was more efficient giving rise to a wider and slower front movement. However, all numerical results produced with second order schemes can be considered qualitatively correct. Even though the beginning of the transient is not completely reproduced by the numerical model, the results show good agreement for the final steady state condition (figures 6a and 6b). Thus it seems that the model predicts mixing region MR3 more or less correctly, but underestimates mixing in MR4.



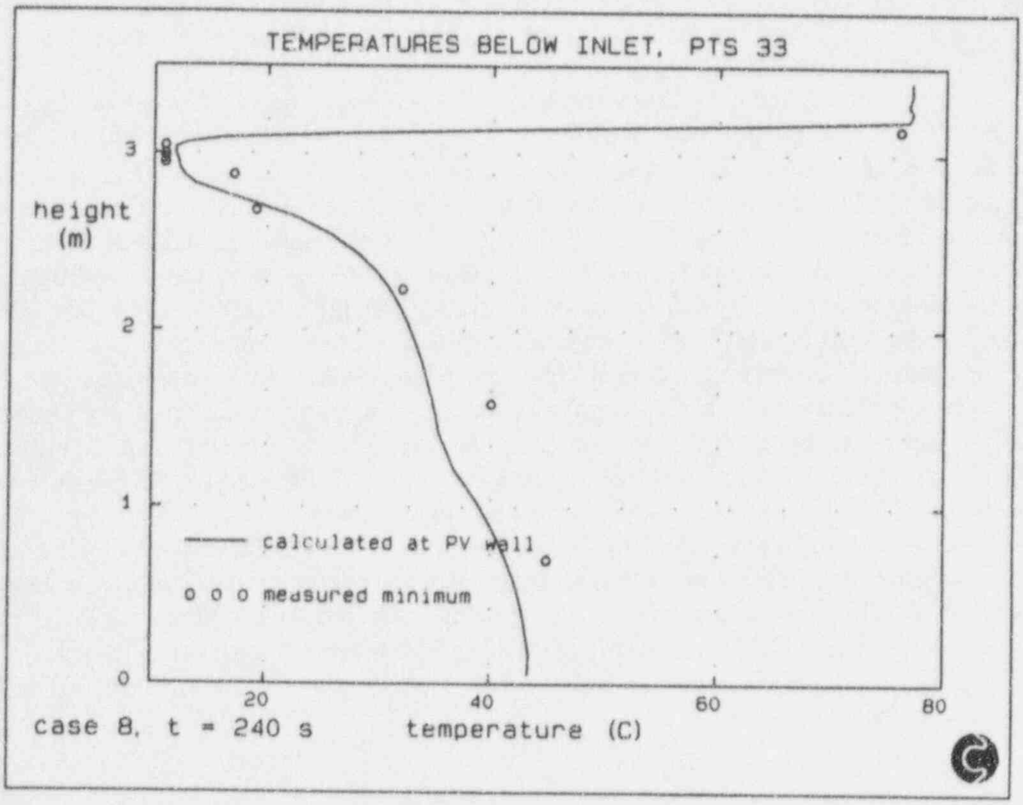


Figure 6a and 6b. Measured and calculated temperatures below inlet, t=240s, tests 22 and 33

3.5.1 Influence of parameters

In buoyant flow conditions the momentum equations and the equations of scalars affecting the density are strongly coupled. Therefore small changes in the solution procedure might have a large effect on the final result. In the results the effect of numerical diffusion was found to be very distinct in the simulations performed with the 1st order discretisation scheme. A shorter time step length gave only weakly improved results. It seems that the usage of 2nd order schemes for all solved quantities is very important in coarse grid buoyant transient flow simulations with sharp interfaces.

The results were found not to be very sensitive to grid density. Although the number of cells in the radial direction in the downcomer has an immediate effect on the wall boundary conditions, the velocity and turbulence fields were only slightly affected by the radial grid density. In the simulations of experiment 33 the length of cold leg piping included in the model was varied, but no changes in the results were noticed. The choice of the value for the turbulent Prandtl/Schmidt number (0.5 or 0.8) for the scalars did not have a large effect on the result; reference 14 recommends the value 0.5 for planar plumes. Neither did the turbulence intensity at inlet affect the results.

3.5.2 Discussion

There are possible sources of errors both in the experiments and the numerical results. In the experiments some information is lost as the thermocouples are located in a relatively coarse grid. The symmetry assumption in the numerical model is an idealization, that underestimates mixing. The setting of the temperature history at model inlet is approximative; more accurate boundary conditions would require an extension of the model to include the HPI nozzle area.

However, once again it has been confirmed that std k-ε turbulence model is not very good for strongly buoyant flows. The reason is the isotropy assumption and assumed applicability of the Reynolds analogy. More sophisticated turbulence models, such as Algebraic Stress Models (ASM) and Reynolds STress Models (RSTM) are expected to perform considerably better. The wider applicability and more general formulation of RSTM makes it more attractive, especially while the saving of computer resources of the simpler ASM compared to RSTM is reported to be only 30-50 % [15]. Additionally it has been claimed, that the ASM expressions are rather complex and difficult to implement in numerical procedures, and so the advantages over RSTM models are lessened or even negated [16].

Nevertheless the results from the simulations were found to be qualitatively correct. They show that the model is capable of predicting the mixing in the loop nozzle area, but not completely the plume decay lower in the downcomer. Although the results from these model validation simulations were not exactly correct they do not reject the usage of the numerical tool in boron mixing studies in natural circulation conditions. The errors noticed are conservative from mixing point of view as mixing was underestimated. Additionally, in many "interesting" (from boron mixing point of view) natural circulation conditions the flows are not as buoyant as in the PTS experiments ($Fr \geq 1$), which decreases the errors.

4 NATURAL CIRCULATION CASE

4.1 Modelled event

In the modelled event the reactor is assumed to be in cold shut-down. The residual heat from the reactor is removed by one steam generator and all other loops are stagnant. The natural circulation flow rate in the working loop is set according to an unrealistic high residual power of 0.7 %, which gives a mass flow rate of 50 kg/s. The coolant density in the cold leg is ρ = 990 kg/s and flow velocity v_{CL} = 0.26 m/s. The governing dimensionless parameters have the values $Re_{CL} \approx 10^5$ and $Fr_{CL} \approx 1$. A diluted slug is assumed to enter the downcomer from the same loop by suddenly changing the boric acid content of the coolant from 12.5 g/kg to zero. Finally, when the whole content of the loop (16 m^3) has been injected to the downcomer, the boric acid concentration is given back its original value 12.5 g/kg. The corresponding boron concentration boundary condition in the cold leg nozzle (model inlet) is shown in figure 7. The event is assumed to be fully isothermal all the time i.e the temperature of the slug is the same as the downcomer temperature.

4.2 Numerical model

After completition of the model validation against PTS experiments the numerical model for the real reactor was extended for buoyant flows. The gravity force, density dependence on temperature and boron concentration $\rho(T,C_b)$ and the buoyancy terms in the equations for k and ε were activated. The buoyancy extended model was taken in use in some introductory natural circulation cases. The grid used in the reported case is shown in figure 8. The time step length used in the transient part of the simulation was Δt = 0.25 s.

Figure 7. Boric acid concentration boundary condition

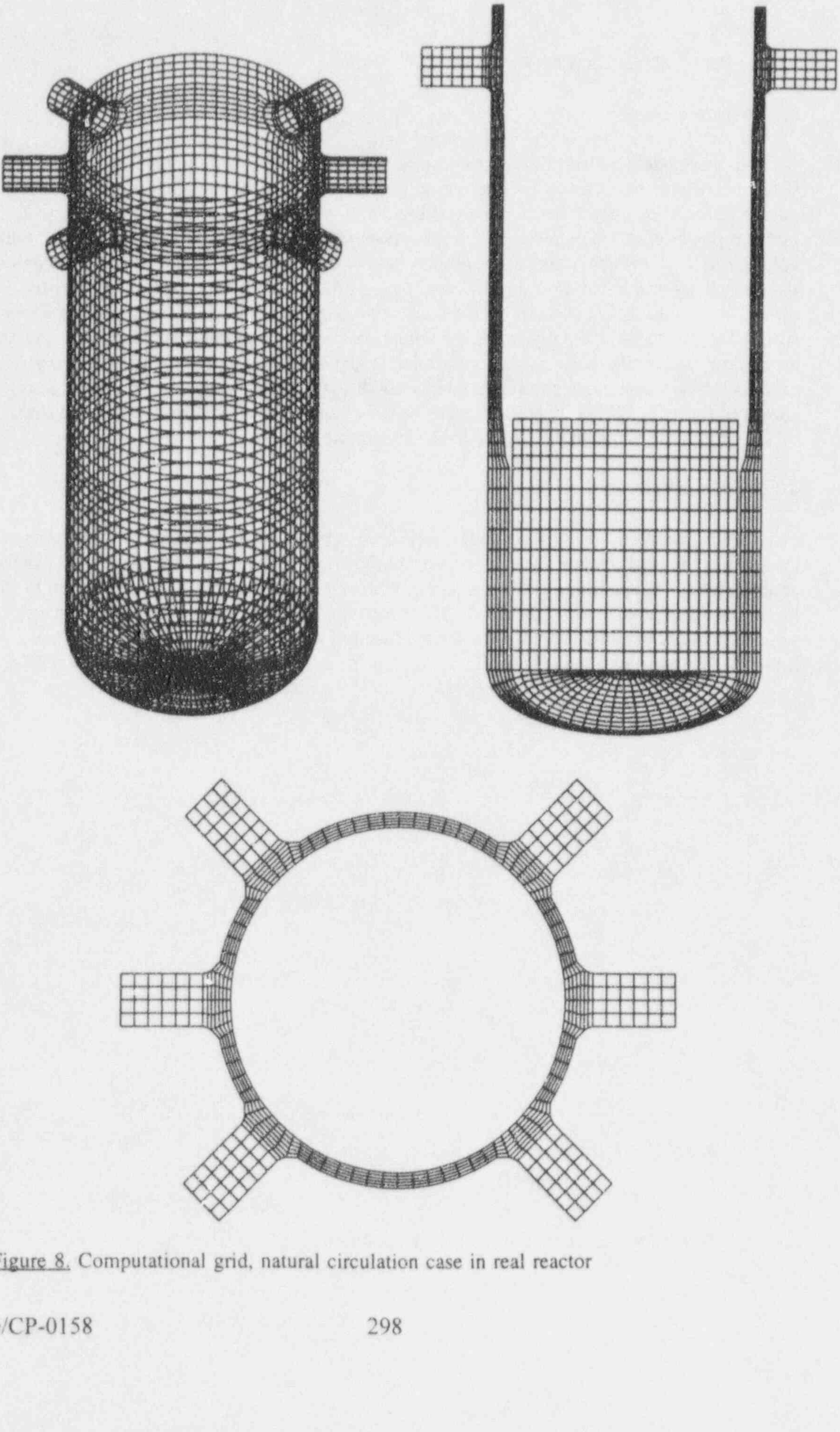


Figure 8. Computational grid, natural circulation case in real reactor

4.3 Results

The simulation of the modelled event was started with a steady state calculation for the initial conditions. The resulting flow field in the downcomer is shown in appendix 2. After achieving the steady state the transient part of the event was started. Resulting flow and boric acid fields at some points of time are shown in appendix 2.

At the beginning of the transient when the lighter slug enters the downcomer stratified conditions develop instantaneously. The downcomer is filled up with the slug and finally all of the pure coolant is accumulated in the downcomer. Almost no mixing has occured so far. Later on, when denser coolant again begins to enter the downcomer a buoyant plume develops in the downcomer. The plume penetrates the layer of diluted coolant rapidly; some "swinging" of the plume is noticed before the final vertical plume develops. Diluted coolant is continuosly mixed to the plume reducing the amount of pure water accumulated in the downcomer. The boron content of the plume is about the half when arriving at the bottom of the downcomer; the lowest boric acid concentration at core inlet was 8 g/kg. The amount of pure water accumulated in the downcomer decreases all the time and at the end of the simulation ($t = 720$ s) only a thin layer of pure water is left in the downcomer.

The results from the simulation show that even a small density difference ($\Delta\rho/\rho$=0.5%) is enough to control the behavior of the slug in the downcomer in natural circulation conditions. Larger density differences are expected to give rise to a similar slug behavior. However, the density difference purely caused by different boron concentrations might in some scenarios be compensated by temperature differences.

5 FURTHER WORK

Further work for studying boron mixing in Loviisa NPP will consist of a comprehensive study of different scenarios with natural circulation restart in the diluted loop in shut-down and accidental conditions. In these cases the HPI, possible flow to leakage and the density differences are expected to produce complex flow patterns that promote mixing and might completely prevent the unmixed transport of the slug to the core. The biggest problem of these scenarios will be the setting of boundary conditions; they can be got for example from RELAP analyses, but inevitably a wide series of simulations will be needed to study the broad range of scenarios. Further work on the pump driven cases is not considered absolutely necessary at this stage, but some confirmatory simulations are planned to be performed.

6 SUMMARY AND CONCLUSIONS

The extension of the numerical boron mixing study program for Loviisa NPP to natural circulation conditions was presumed to require a validation of the buoyancy extended numerical model. The PTS mixing experiments performed at IVO in the mid 80s were found suitable for serving as experimental reference data. In this paper the results of the numerical simulation of IVO PTS mixing experiments 22, 33 and 47 have been reported. Additionallly the results from one of the introductory natural circulation cases for the real reactor were given.

The results from the PTS simulations brought up the importance of using higher order discretization schemes for the convection terms in <u>all</u> equations. Numerical effects seem to be emphasized in buoyant flow conditions, were all equations are strongly coupled. However, an underestimation of mixing was reported in the beginning of the transient in all cases. Evidently the standard k-ε turbulence model is not fully capable of predicting the plume behavior during the initial part of the transient. The final state was reproduced quite well by the model in all simulated experiments. The results showed more or less insensitivity to grid density and some other minor parameters.

In the natural circulation case reported the density difference purely caused by the difference in boron concentration ($\Delta\rho/\rho = 0.5\%$) seemed to be enough to control the behavior of the slug and to ensure efficient mixing. The lighter slug was accumulated as a layer in the upper parts of the downcomer and later on, when denser coolant entered the downcomer, the slug was mixed little by little into the resulting plume. Thus only well mixed coolant was reported to enter the core in this case.

In the inhomogenous boron dilution problem mixing might serve as an inherent protection mechanism. However, the results from the numerical work performed until now show that mixing is case and plant specific; the high and open downcomer geometry of VVER-440 together with a special design of the reactor bottom seems to be advantageous from mixing point of view both in forced and buoyant flow conditions. Accordingly many of the possible scenarios have to be studied separately for different reactor designs. Properly validated numerical CFD models seem to serve as tools for this work.

REFERENCES

[1] Jacobson, S. Some local dilution transients in a pressurized water reactor. Linköping University, Diploma thesis no. 171, Sweden, March 1989.

[2] Hyvärinen, J. The inherent boron dilution mechanism in pressurized water reactors. Nuclear Engineering and Design, **145** (1993) 227-240.

[3] Alvarez, D. et al. Three dimensional calculations and experimental investigations of primary coolant flow in a 900 MW PWR vessel. Fifth International Topical Mtg. on Reactor Thermal Hydraulics (NURETH-5), Salt Lake City, September 1992, Vol. II, pp. 586-592.

[4] Jacobson, S. Risk evaluation of local dilution transients in a pressurized water reactor. Linköping University, Studies in Science and Technology, Dissertations No. 275, 1992.

[5] Gango, P. Numerical boron mixing studies for Loviisa nuclear power plant. Seventh International Meeting on Nuclear Reactor Thermal-Hydraulics (NURETH-7), Saratoga Springs, NY, USA, September 1995.

[6] Gango, P. Loviisa 1, Development of CFD -model for the reactor downcomer. IVO International Ltd, working report LO1-K859-136, 12 September 1994 (in Finnish).

[7] Tuomisto, H. Thermal-hydraulics of the Loviisa reactor pressure vessel overcooling transients. Imatran Voima Oy, Research Reports, IVO-A-01/87, Helsinki, June 1987.

[8] Spalding, D.B. PHOENICS Beginner's Guide.

[9] Rodi, W. The simulation of turbulence in practical flow calculation. In: Proceedings of Basel World CFD User Days 1992. Basel 24-28 May 1992.

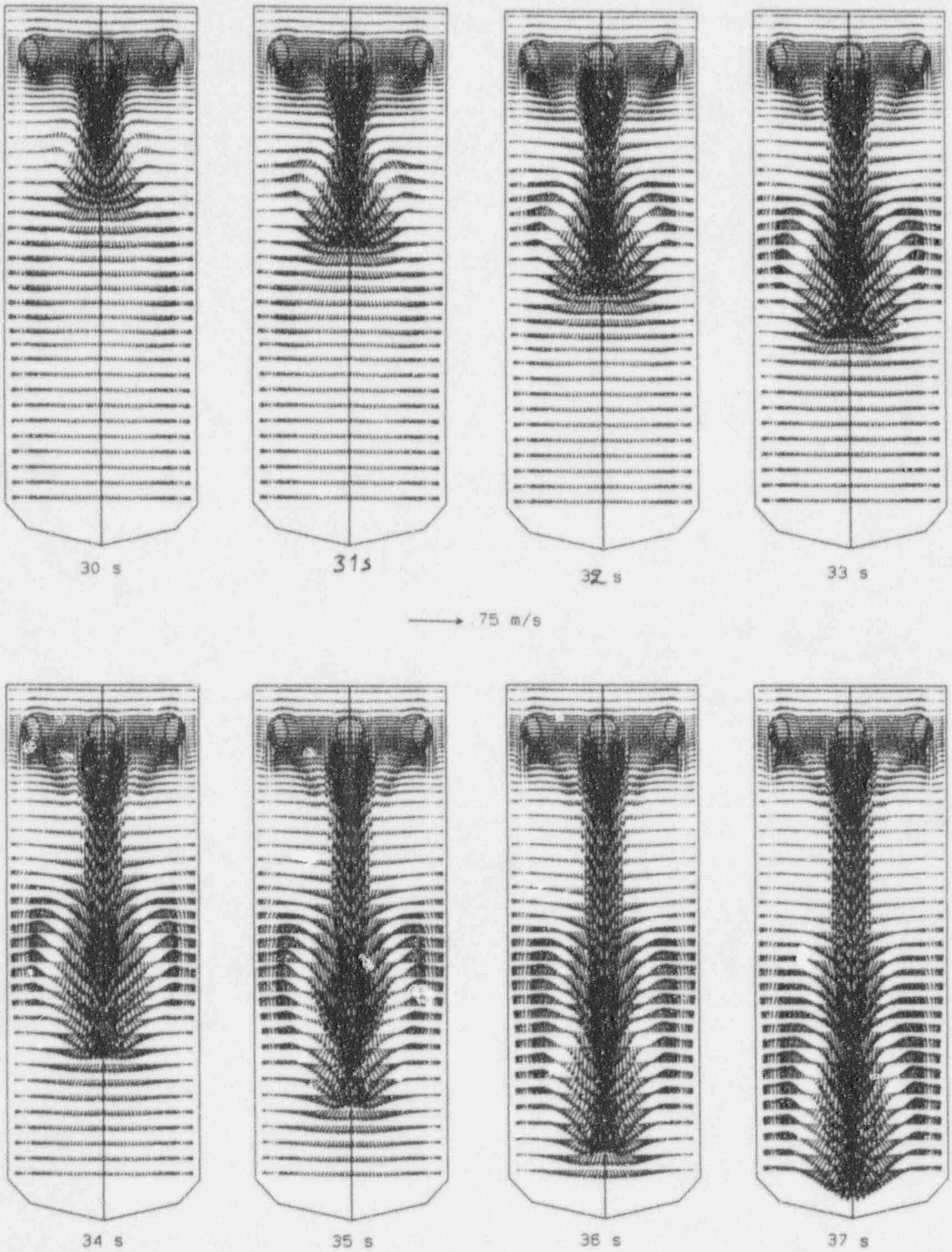
[10] Mustonen, P. Fluid and thermal mixing tests of the Loviisa pressure vessel downcomer. Imatran Voima Oy, Hydraulic Laboratory, Test Report RLB 322, March 1984.

[11] Theofanous, T.G. et al. Decay of buoyancy driven stratified layers with applications to PTS, Purdue University, NUREG/CR-3700, May 1984.

[12] Wolf, L. et al. Overview of HDR large scale PTS thermal mixing experiments and analyses with 3-D codes and engineering models, Transactions of 8th International Conference on Structural Mechanics in Reactor Technology (SMiRT-8), Brussels, Belgium, Aug 19-23, 1985 vol. F1, F2 pp. 359-365

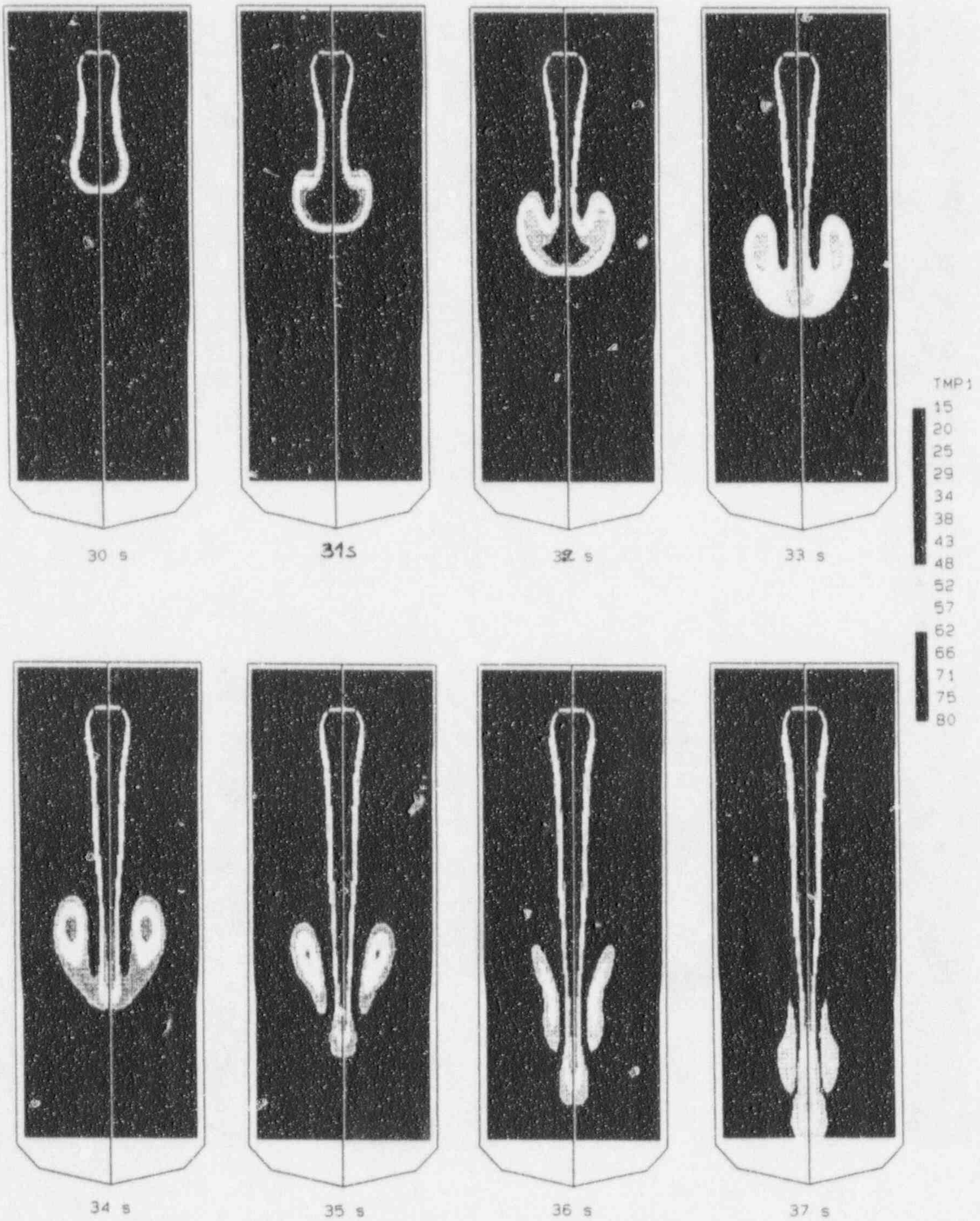
[13] Tuomisto, H. Experiments and analyses of thermal mixing and stratification during overcooling accidents in a pressurized water reactor, 24th ASME/AIChE National Heat Transfer Conference, Pittsburgh, PA, August 1987.

[14] Chen, C.J., Rodi, W. Vertical turbulent buoyant jets, A review of experimental data. Oxford, Pergamon Press, 1980.
[15] Launder, B.E. Current capabilities for modelling turbulence in industrial flows. Applied Scientific Research, **48** (1991) 247-269.
[16] Malin, M.R., Younis, B.A. Calculation of turbulent buoyant plumes with a Reynolds stress and heat flux transport closure. Int. J. Heat and Mass Transfer, **33** (1990) No. 10 2247-2264.



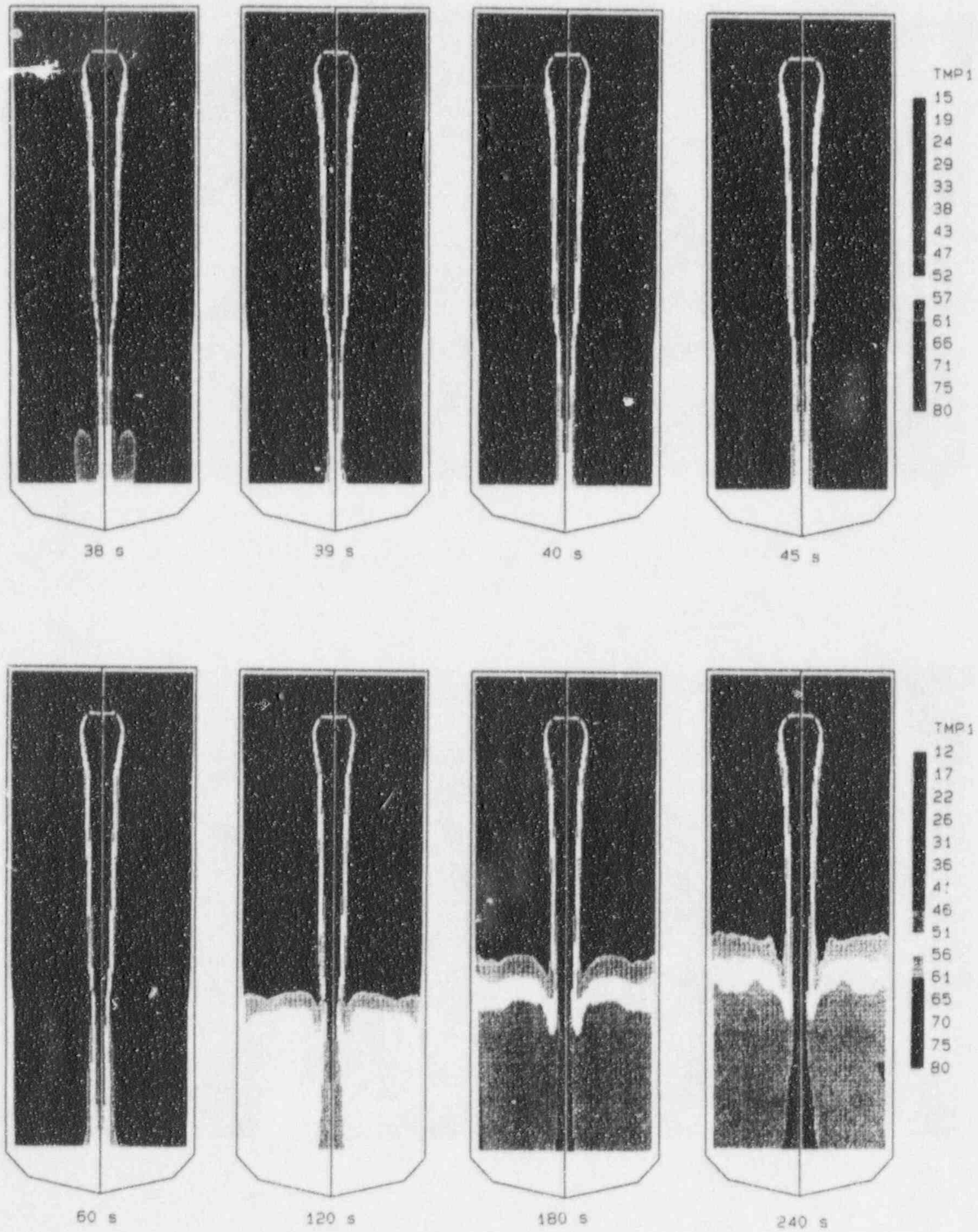
PTS 33, CASE 8

Velocity field in the downcomer, t = 30 - 37 s



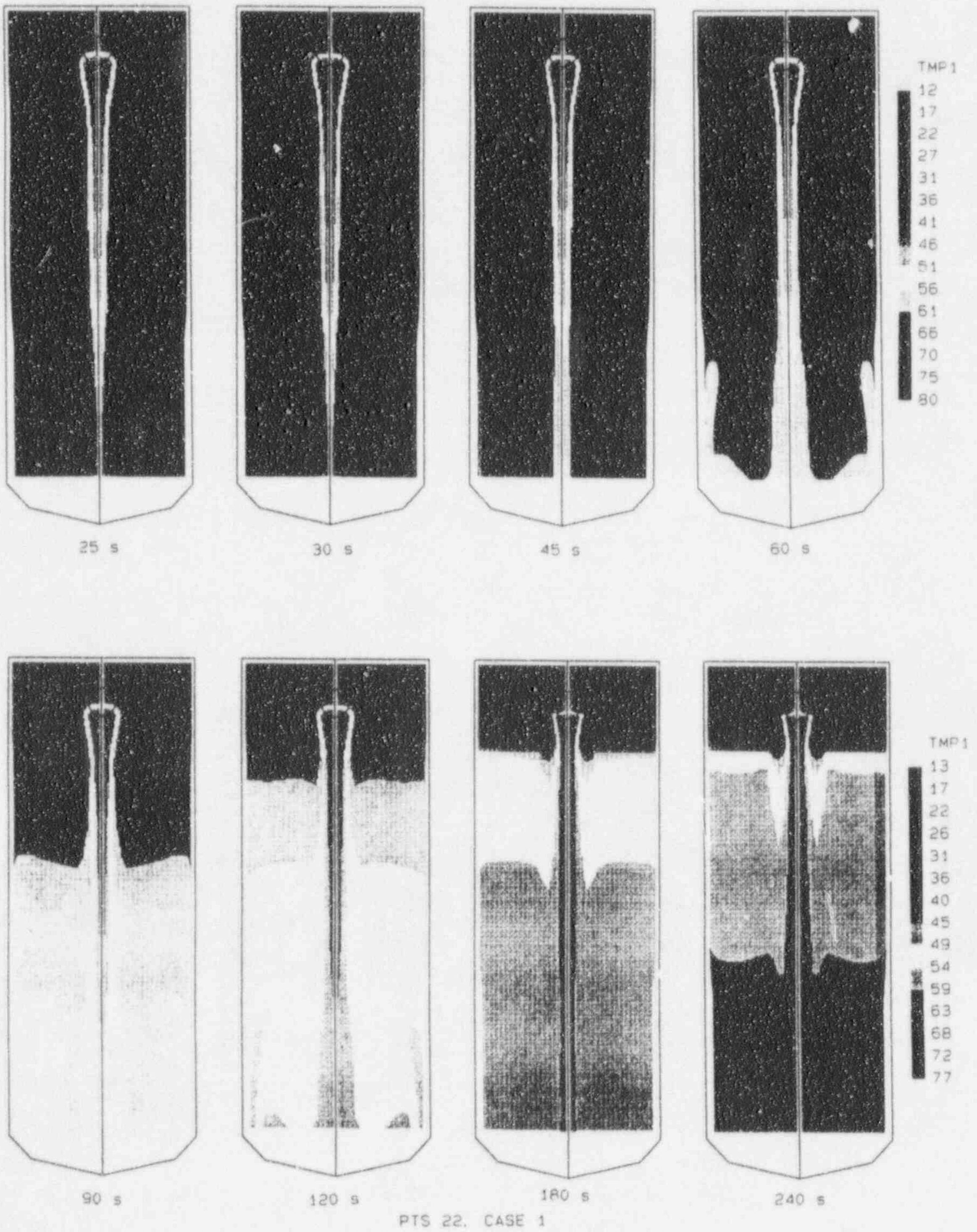
FTS 33, CASE 8

Temperature field in the downcomer, t = 30 - 37 s

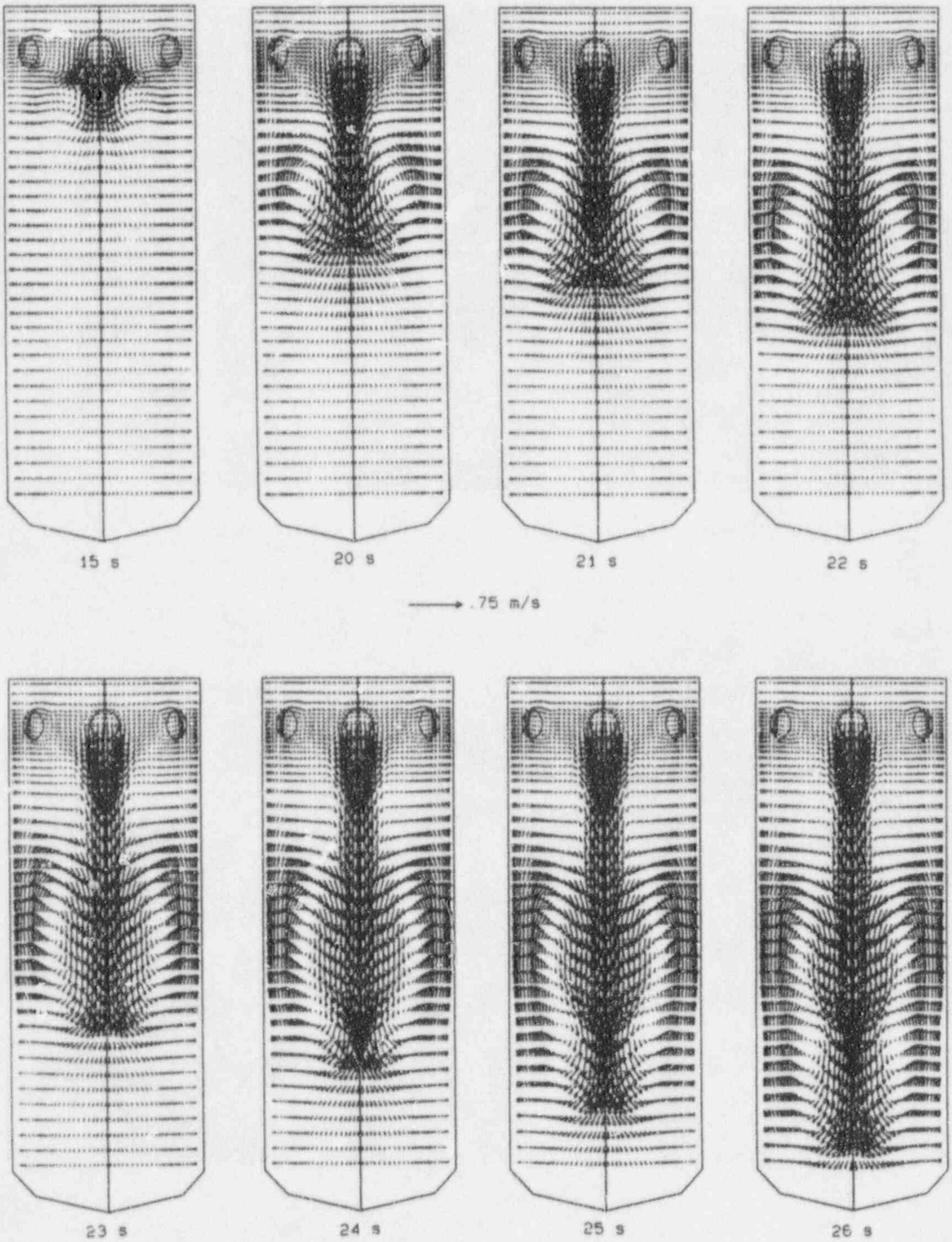


PTS 33, CASE 8

Temperature field in the downcomer, t = 38 - 240 s

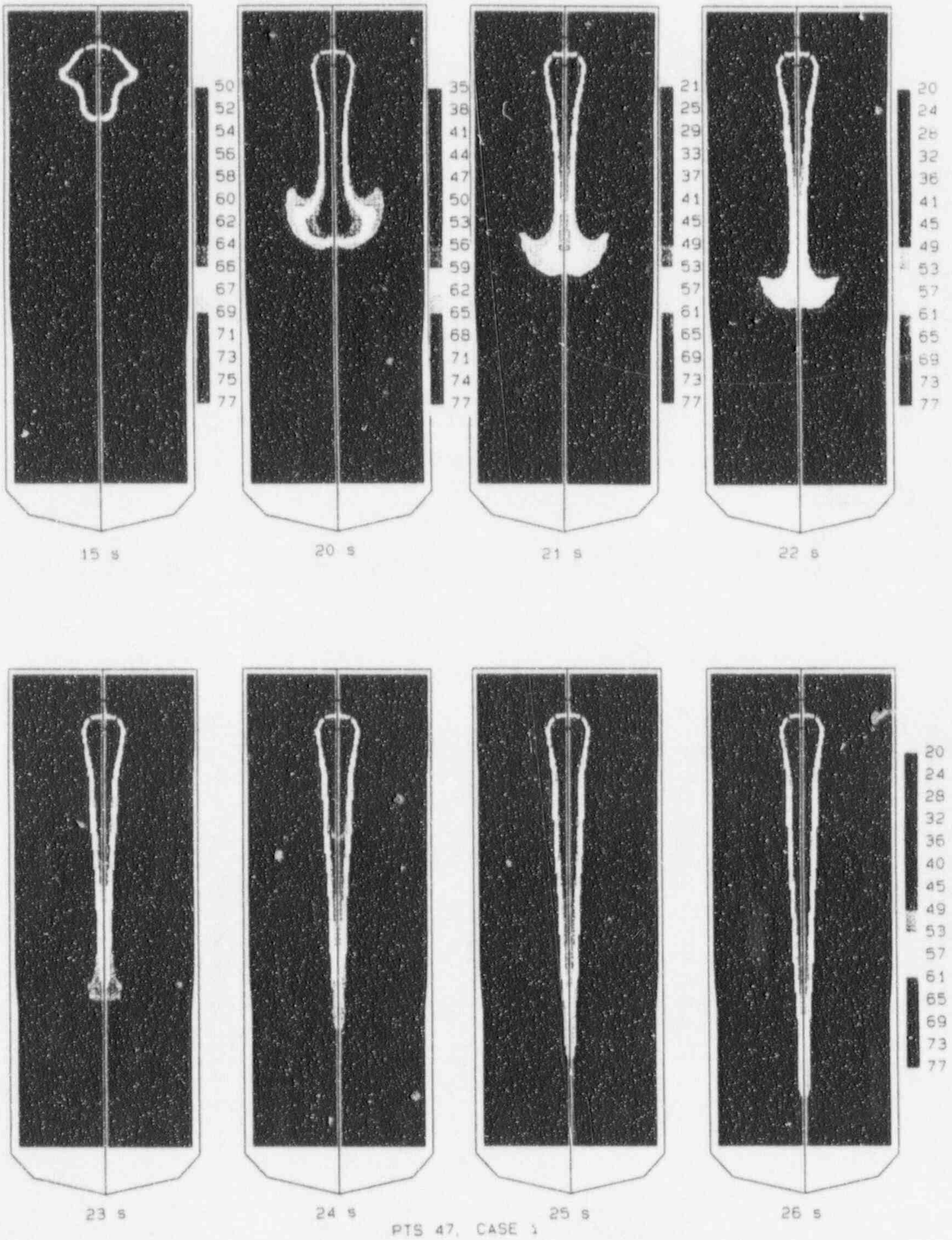


Temperature field in the downcomer, t = 25 - 240 s

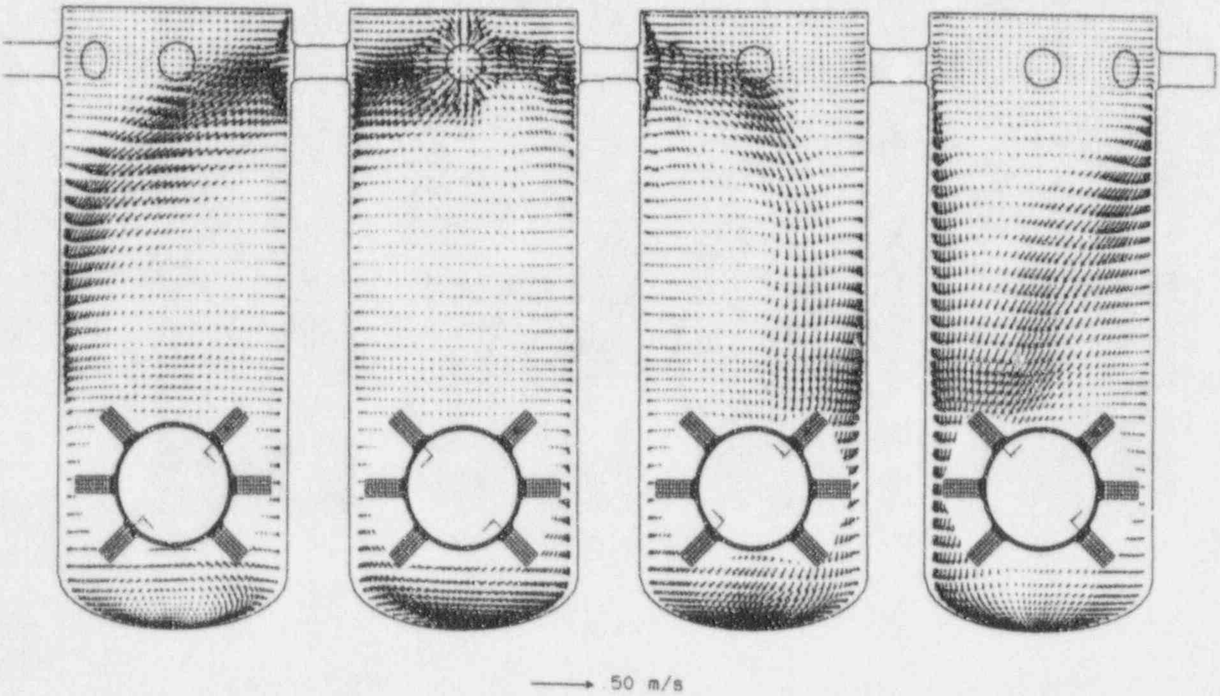


PTS 47, CASE 1

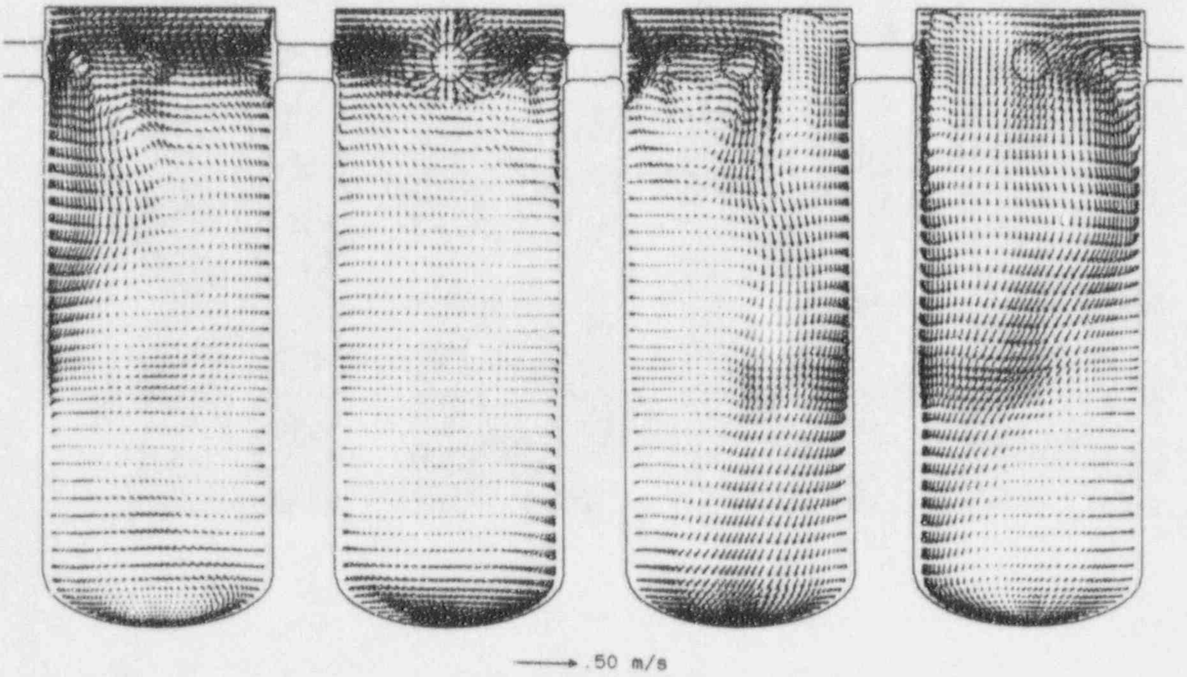
Velocity field in the downcomer, t = 15 - 26 s



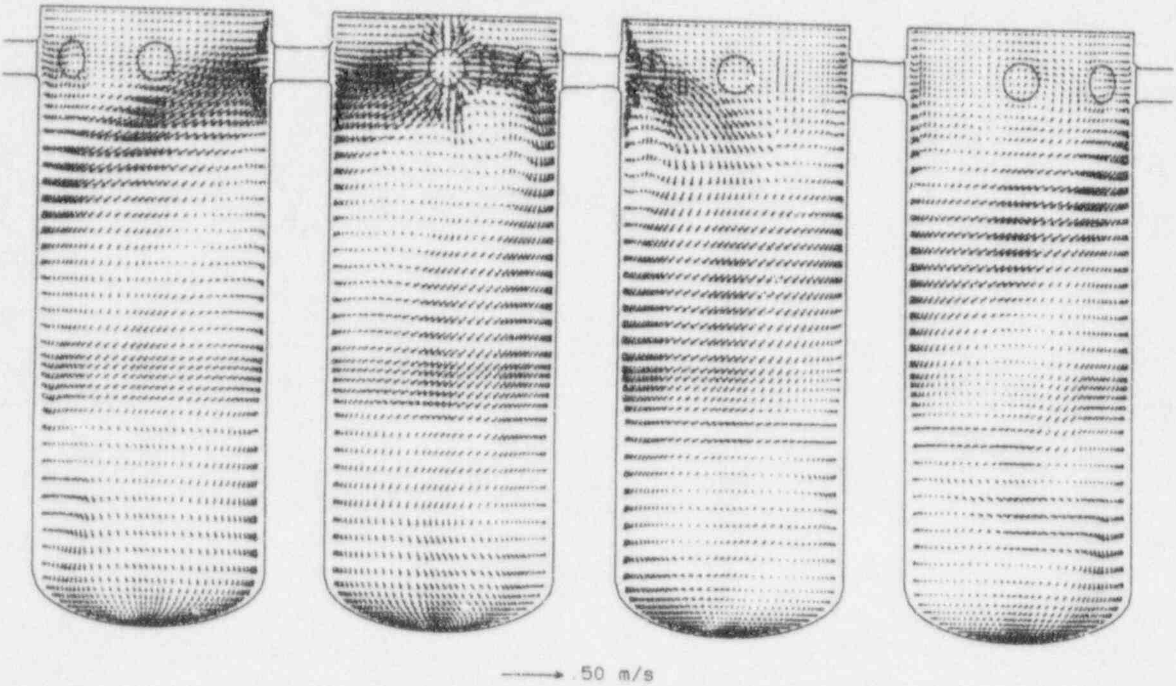
Temperature field in the downcomer, t = 15 - 26 s



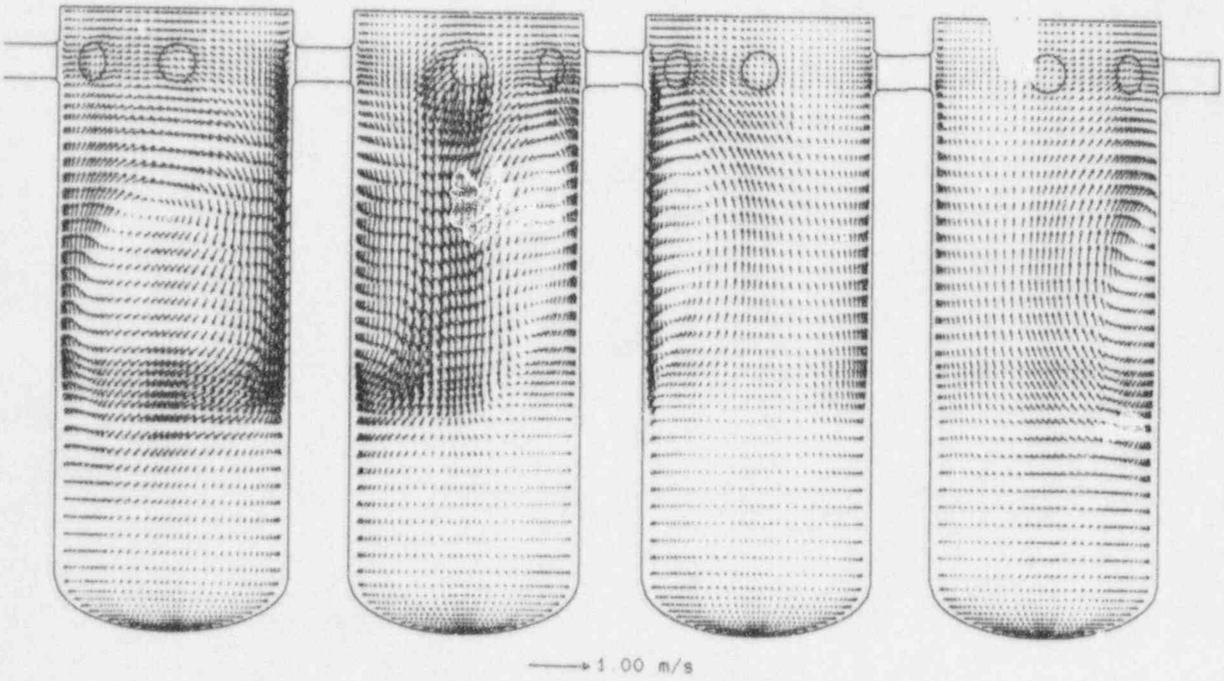
LO1, NATURAL CIRC., CASE 4, steady



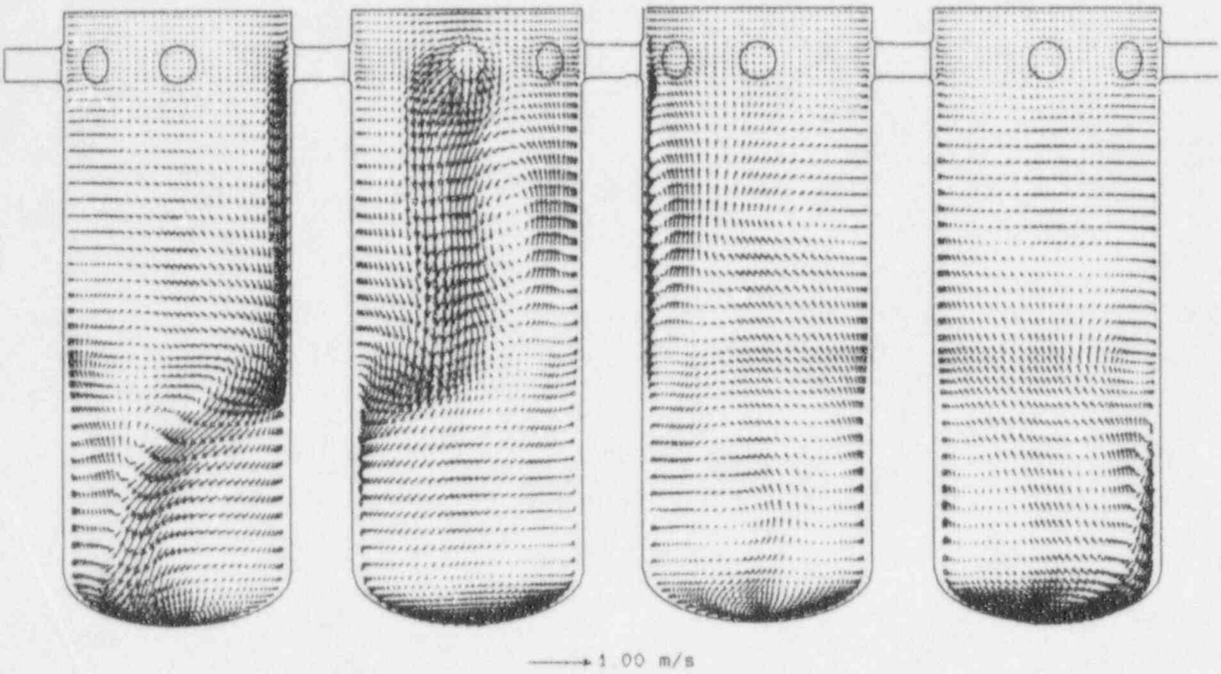
LO1, NATURAL CIRC., CASE 4, 30s



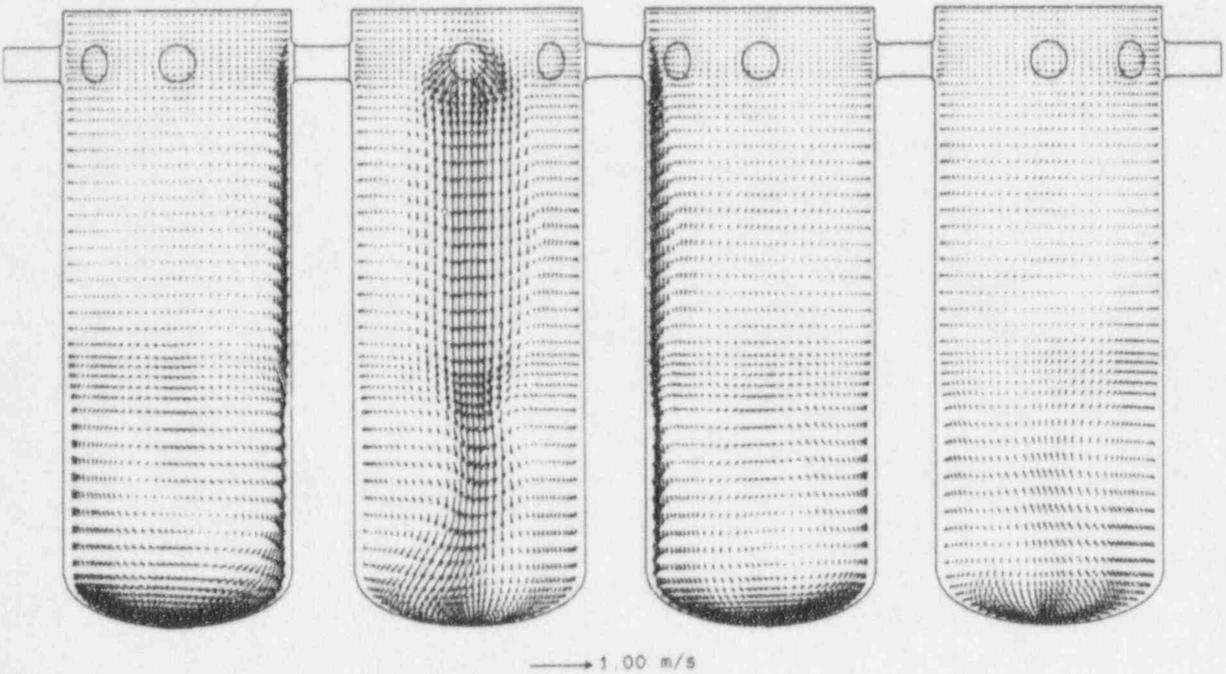
LO1, NATURAL CIRC., CASE 4, 250s



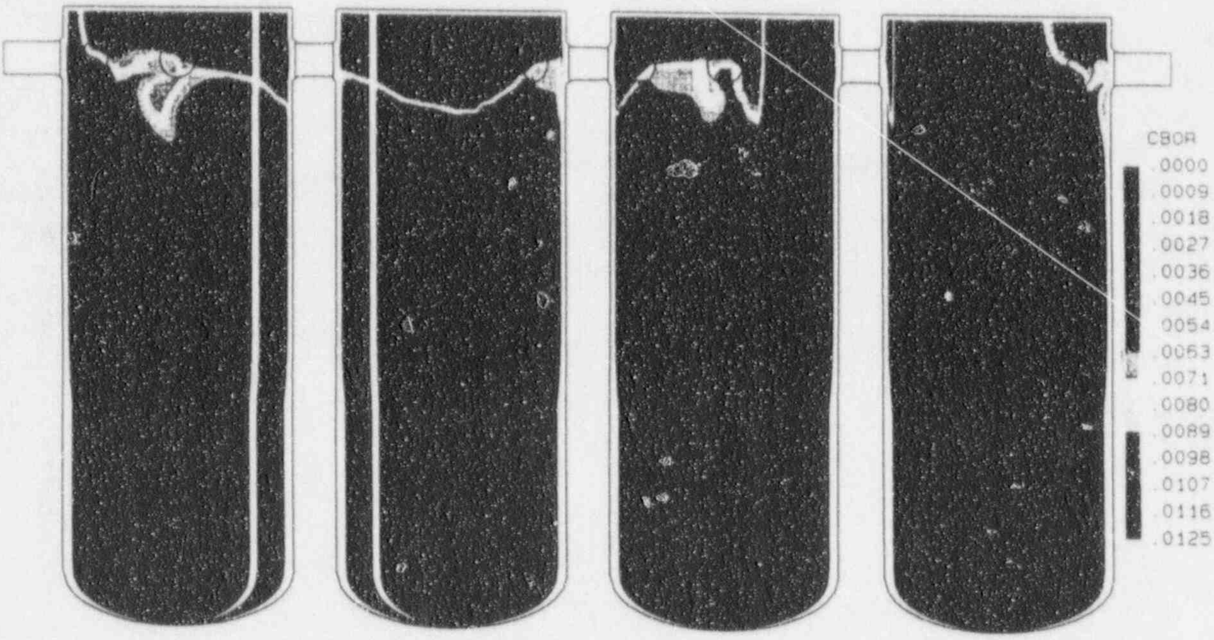
LO1, NATURAL CIRC., CASE 4, 345s



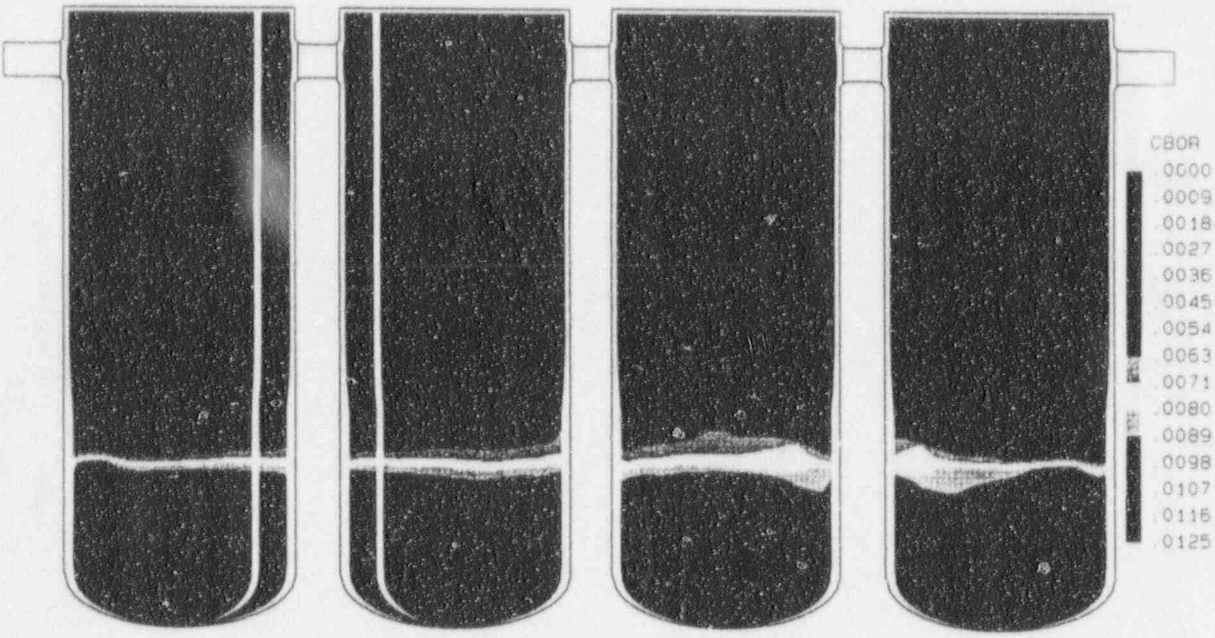
L01. NATURAL CIRC., CASE 4, 400s



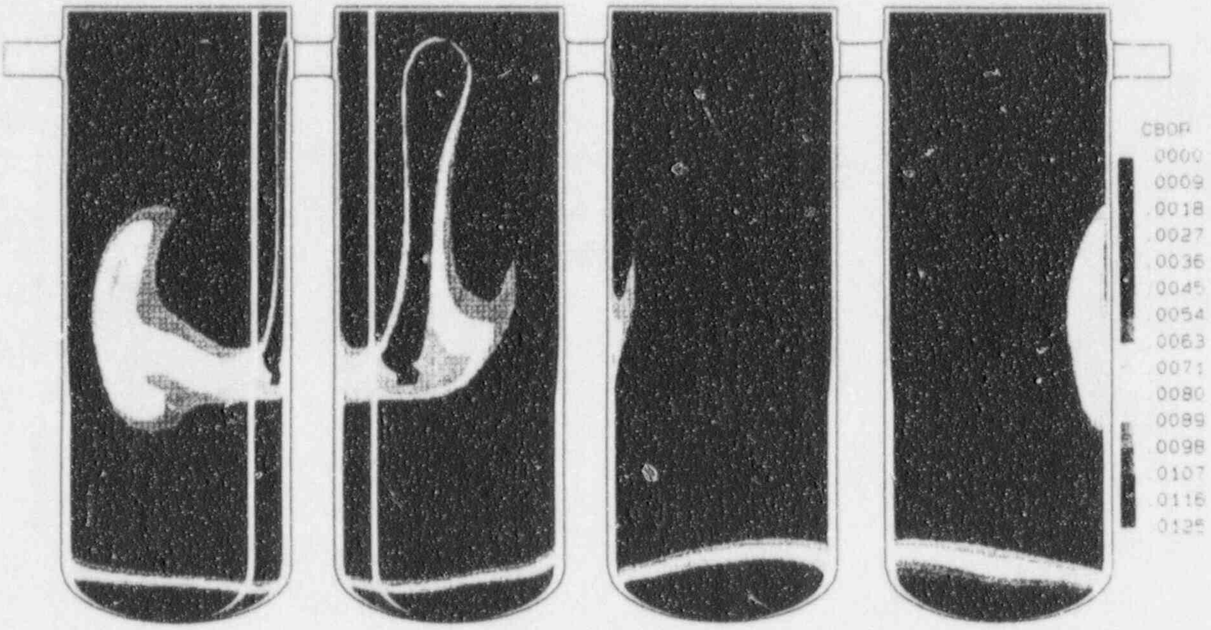
L01. NATURAL CIRC., CASE 4, 660s



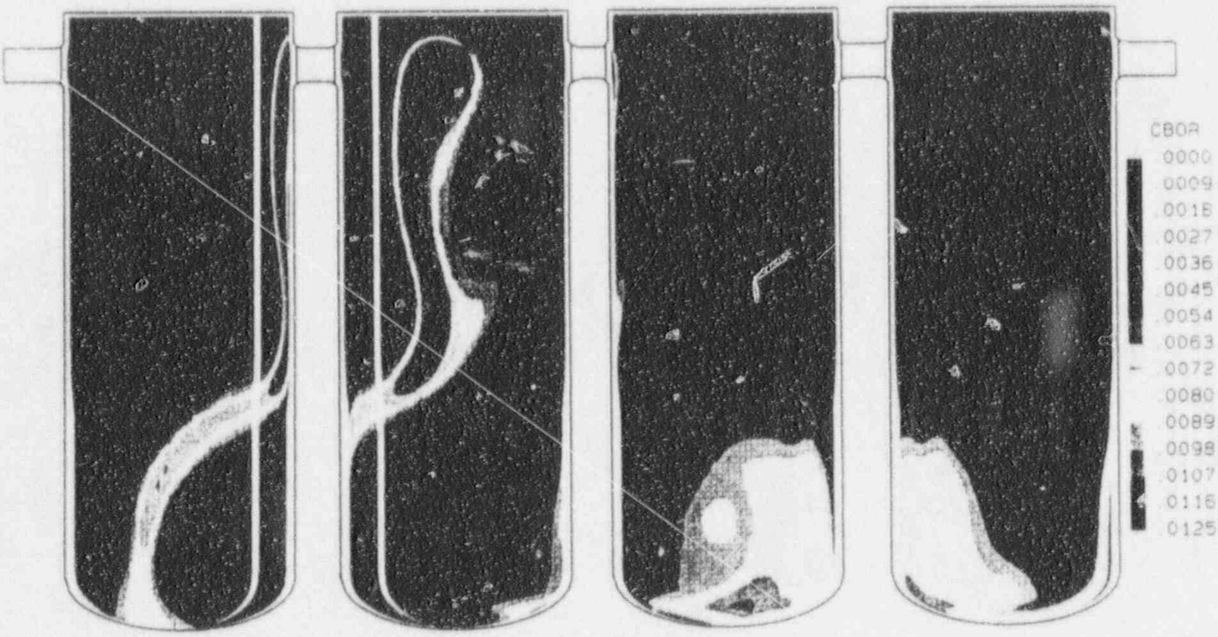
LO1. NATURAL CIRC., CASE 4. 30s



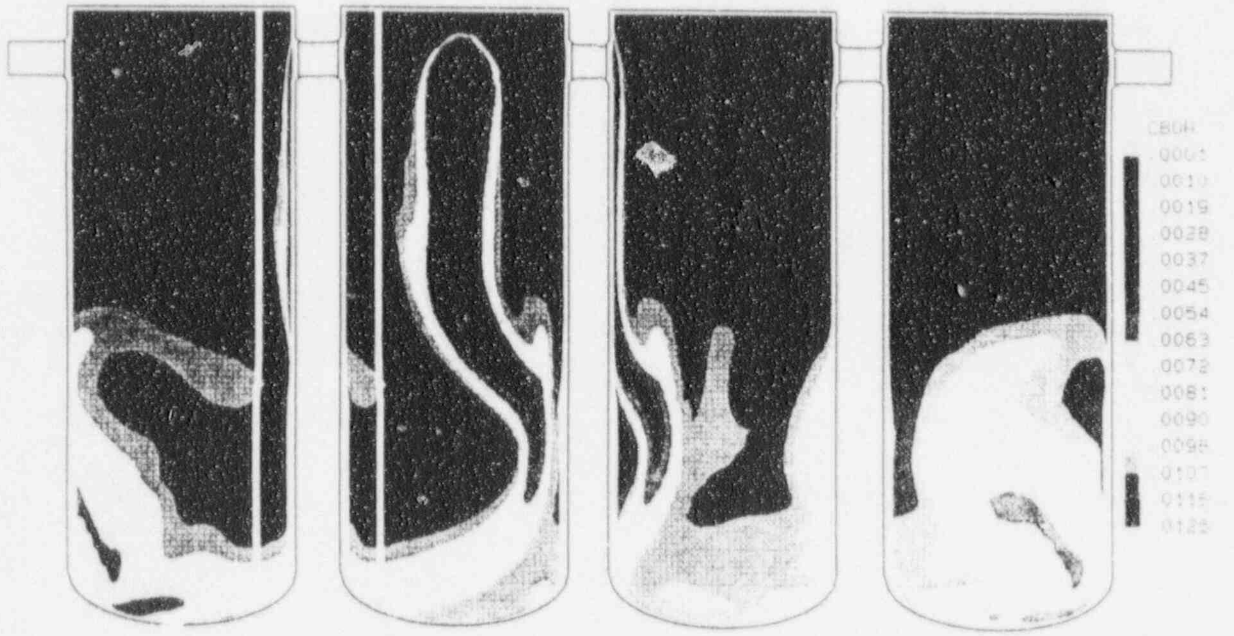
LO1. NATURAL CIRC., CASE 4. 250s



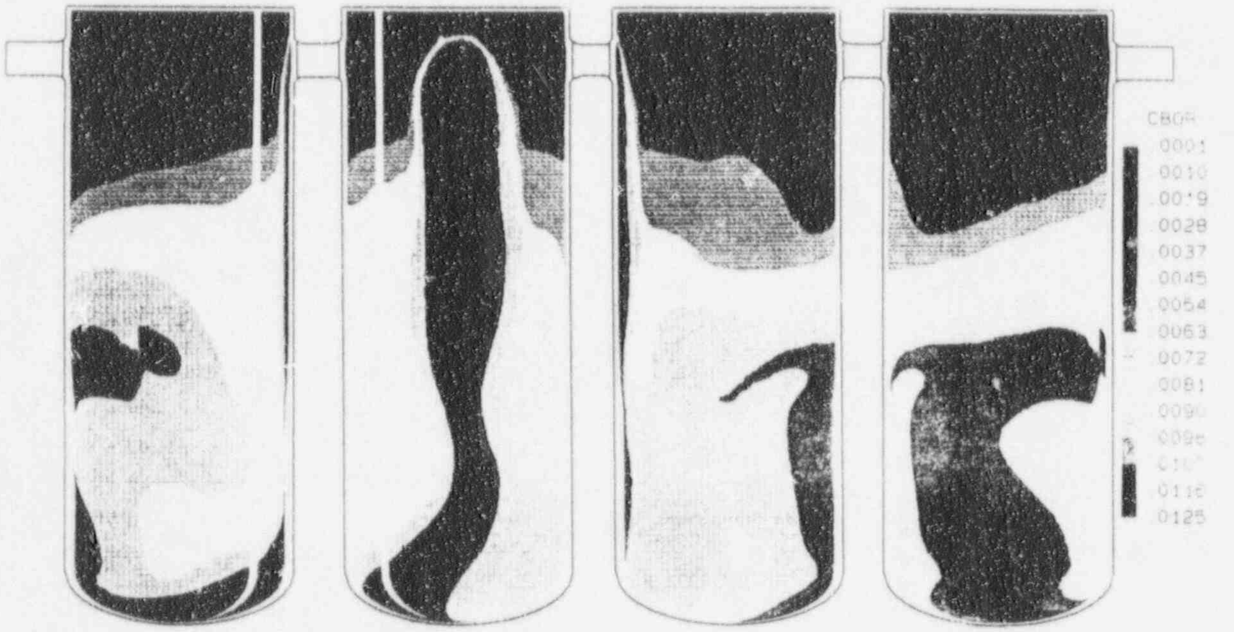
LO1, NATURAL CIRC., CASE 4, 145s



LO1, NATURAL CIRC., CASE 4, 400s



LO1.NATURAL CIRC . CASE 4.510s



LO1.NATURAL CIRC . CASE 4.660s

Potential for Boron Dilution During Small-Break LOCAs in PWRs *

H.P. Nourbakhsh and Z. Cheng
Safety and Risk Evaluation Division
Department of Advanced Technology
Brookhaven National Laboratory
Upton, N.Y. 11973

ABSTRACT

This paper documents the results of a scoping study of boron dilution and mixing phenomena during small break loss of coolant accidents (LOCAs) in pressurized water reactors (PWRs). Boron free condensate can accumulate in the cold leg loop seals when the reactor is operating in a reflux/boiler condenser mode. A problem may occur when the subsequent change in flow conditions such as loop seal clearing or re-establishment of natural circulation flow drive the diluted water in the loop seals into the reactor core without sufficient mixing with the highly borated water in the reactor downcomer and lower plenum. The resulting low boron concentration coolant entering the core may cause a power excursion leading to fuel failure. The mixing processes associated with a slow moving stream of diluted water through the loop seal to the core were examined in this report. A bounding evaluation of the range of boron concentration entering the core during a small break LOCA in a typical Westinghouse-designed, four-loop plant is also presented in this report.

1 INTRODUCTION

Boric Acid is used as a soluble neutron absorber for long-term reactivity control in pressurized water reactors (PWRs). A reduction in the boron concentration (i.e., boron dilution) in the core region may result in a reactivity accident with the potential to cause a power excursion and fuel damage.

Boron dilution events have always been of concern in PWRs. The required safety analyses for PWRs include events in which positive reactivity is added to the core due to inadvertent boron dilution. For these boron dilution events, it is assumed that the dilution flow is injected to a continuously flowing reactor coolant which leads to homogenous dilution of all the primary coolant inventory. For homogenous dilution events a large volume of diluted water is required and the boron concentration in the core changes slowly which leaves plenty of time for the identification of the problem and operator intervention.

The accident at the Chernobyl Nuclear Power Plant provided an impetus for renewed interest in reactivity accidents in PWRs. Studies by several European organizations (mainly in France, Sweden and Finland) have postulated numerous new boron-dilution-induced reactivity transients in PWRs. These scenarios assume accumulation of diluted water in a stagnant part of the reactor coolant system (RCS) during a period when there is very little circulation. Due to subsequent change in flow conditions such as start of a reactor coolant pump (RCP), the depleted zone will be transported to and through the reactor core and cause a reactivity transient. Most of these heterogeneous (local) dilution events assume the addition of boron-free coolant from an external source, such as the chemical and volume control system (CVCS), erroneously diluted accumulator, reactor coolant pump seal injections or a reversed steam generator leak.[1]

Recently, a mechanism has been identified that leads to heterogeneity in boron concentration without any external source of diluted water.[2-3] This dilution mechanism is via accumulation of

*Work performed under the auspices of the U.S. Nuclear Regulatory Commission.

boron free condensate in the cold leg loop seals due to reflux/boiler-condenser mode operation during certain accidents, such as small break loss of coolant accidents (LOCAs). The subsequent change in flow conditions, such as loop seal clearing or re-establishment of natural circulation flow, may provide an effective mechanism to drive the slug of diluted water into the core. However, the buoyancy and turbulent mixing process along the way from the loop seal to the core may sufficiently increase the boron concentration of the diluted stream to prevent a power excursion leading to fuel failure.

The general objective of the work presented in this paper is to improve the understanding of the boron dilution mechanism and the mixing phenomena during small break LOCAs in PWRs.

The mixing processes associated with a slow moving stream of diluted water through the loop seal to the core are examined in this paper. A preliminary evaluation of the range of boron concentration entering the core during a small break LOCA in a typical PWR is also presented in this paper. It must be recognized that the focus of this study is limited to a Westinghouse-designed, four-loop plant. The reactor design studied is the RESAR-3S. No attempt is made to extrapolate the results to other Westinghouse designs or to Combustion Engineering or Babcock & Wilcox reactor designs.

2 THE THERMAL HYDRAULICS OF SMALL-BREAK LOCAs RELEVANT TO BORON DILUTION

There has been significant research activity related to small-break loss-of-coolant accidents (LOCAs) following the accident at Three Mile Island Unit 2 (TMI-2). The experimental and analytical results from these research programs have provided a better understanding of the important physical phenomena relevant to small break loss of coolant accidents.

A small-break LOCA is characterized by slow RCS depressurization rates and low fluid velocities within the reactor coolant system (RCS) as compared to a design basis large-break LOCA. Because of the slow depressurization rate, various phase change/separation phenomena dominate the thermal hydraulic characteristics of small-break LOCAs (SBLOCAs). One aspect of this behavior is the existence of an inherent boron dilution mechanism in the course of SBLOCAs that involves the decay heat removal by phase-separating natural circulation (i.e., reflux/boiler condenser mode operation). The steam that is generated in the core is largely devoid of boric acid. Due to subsequent condensation in steam generators, a portion of boron-free condensate can run down the downflow side of steam generator tubes and accumulate in the loop seals between the steam generator outlet plena and the reactor coolant pumps (RCPs). A problem may occur when the subsequent change in flow condition such as loop seal clearing or re-establishment of natural circulation flow drive the diluted water in the loop seals to the reactor core without sufficient mixing with the highly borated water in the reactor downcomer and lower plenum. The resulting low boron concentration coolant entering the core may result to a power excursion leading to fuel failure. The phenomena and processes of interest to boron dilution issue during small-break LOCAs are summarized in Figure 1.

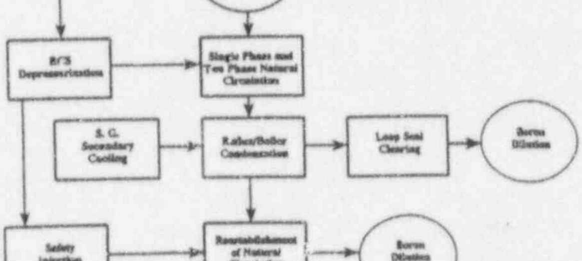


Figure 1 Phenomena and processes of interest to boron dilution issue during small break LOCAs

In this section, a general discussion of the small-break LOCA phenomenology relevant to the boron dilution issue is presented. In particular, this chapter discusses the conditions under which boron dilution occurs. The mixing processes associated with a moving stream of diluted water through the loop seals to the core are discussed in Section 3.

2.1 Characteristics of Small Break LOCA Scenarios

The small-break LOCAs as generally defined, include any break in the PWR pressure boundary with an area less than 4.65×10^{-2} m^2 (0.5 ft^2). This range of break areas encompasses all small lines that penetrate the RCS pressure boundary[4].

The emergency core cooling system (ECCS) is the principal PWR design feature for mitigating the consequences of a small-break LOCA. The purpose of the ECCS is to restore the water inventory in the RCS, thus provide for sufficient core cooling. The major subsystems of a typical US PWR ECCS are the high pressure safety injection (HPSI) system, the safety injection tanks (SITs) or accumulators and the low-pressure safety injection (LPSI) system. Because of the need for heat removal by the steam generators, the auxiliary feed water system is also important for small-break LOCA mitigation.

The major characteristics of the transient response to the small-break LOCAs are similar for all U.S. PWR designs. The magnitude and timing of the physical phenomena as functions of break size depend on plant geometry, break and injection locations, emergency core cooling (ECCS) capacities and equipment failure criteria, all of which differ considerably in various designs. For the present study, the results of TRAC and RELAP calculations for cold leg small break LOCAs in a Westinghouse-designed, four-loop plant (RESAR-3S) reported in Reference [5] are utilized. These calculations are based on the minimum availability of ECCS equipment required by licensing regulations in the United States; particularly, credit is taken for one HPSI and one charging pump. Only one motor-driven auxiliary feedwater pump was assumed to be available. Summaries of pertinent Westinghouse plant data for a four-loop design are shown in Figure 2.

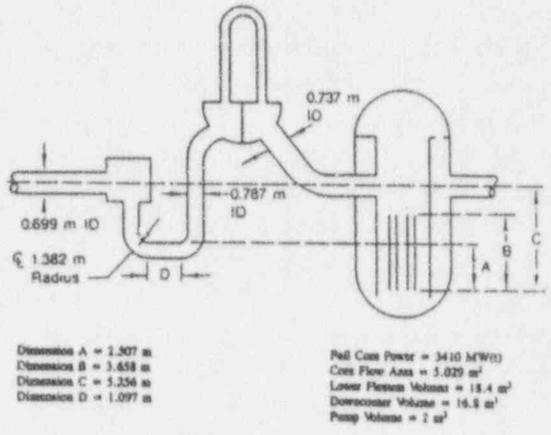


Figure 2 Pertinent data for a typical Westinghouse design Four-Loop Plant (RESAR-3S)

If the break area is large enough (0.95 cm diameter in the reference plant) that the charging pumps cannot maintain the reactor coolant inventory, the RCS will depressurize. The depressurization causes a reactor trip signal, a reactor coolant pump trip signal and a safety injection actuation signal at ~ 12 MPa (~ 1750 Psia). The reactor coolant pumps are stopped to reduce coolant loss through the break. Natural circulation is thus required to provide the heat removal from the core to the steam generators.

The rate of RCS depressurization following high pressure safety injection depends on the break size. Three major classes of small-break LOCAs are commonly identified:[4,6]
(1) breaks that are large enough to depressurize the RCS to the set-point pressure of the accumulators, (2) smaller breaks that lead to a quasi-steady pressure plateau, and (3) even smaller breaks that may lead to RCS repressurization due to injection

via the HPI pumps. The range of break areas in each class depends on the PWR design parameters, including the physical layout of the loops, core power, HPSI pump capacity and accumulator set point pressure[4].

For relatively large small-break LOCAs, the RCS depressurize to the accumulator injection setpoint pressure. For the reference PWR design these breaks have an area of $>2x10^{-3}m^2$ (>2 in diameter break). For breaks in this range, the reactor coolant flow through the break is sufficient to remove core decay heat load without any additional cooling through the steam generators. The pressure drops very rapidly with a short interval pressure plateau above the secondary side safety valve setpoint before loss of natural circulation occurs. A halt in depressurization occurs because the energy removal through the break, which is limited by the critical flow, is less than the core decay heat. The excess energy is removed to the secondary water in the steam generators. However, after break uncovery, the rate of energy removal through the break increases and the RCS continues to depressurize until accumulator injection.

A second class of small-break LOCAs is defined as those in which the net rate of reactor coolant inventory loss is arrested before the RCS depressurizes to the accumulator setpoint pressure. For breaks in this range (1 to 2 inches in diameter for the reference PWR) of the small-break LOCA spectrum, the initial RCS depressurization is similar to that for larger breaks, including rapid initial depressurization and a pressure plateau dictated by secondary-water temperature. However, when the break is uncovered and steam is discharged, only a short period of rapid depressurization occurs. This is followed by very slow depressurization, which continues so long as heat is being removed by the steam generators. For this class of small-break LOCAs, the HPSI pumps and steam generators play important roles: The HPSI pumps provide makeup water to replenish the reactor coolant inventory and steam generators remove heat from the RCS. Either the steam dump system (preferred) or the steam generator power operated relief valves can be used as the mechanism to extend secondary side cooling of the RCS. Where steam generator cooling is not possible, feed and bleed cooling is evaluated.

The third class of small-break LOCAs are those which may repressurize either by loss of the steam generators as a heat sink, by isolation of the break or by HPSI flow exceeding the inventory loss through the break[4]. For the reference PWR plant, these smaller-size small- breaks have an area of less than $5.07x10^{-4}$ m^2 (one inch diameter break). Decay heat in this class of LOCAs is removed almost entirely by the steam generators.

If the break area is large enough, there will be a transition to reflux boiling. Depending on the break size, the coolant level in the RCS may continue to decrease during the period of reflux boiling. Eventually, the RCS pressure will decrease sufficiently that the rate of water injected by ECCS will exceed the rate of flow out of the break and the primary system may be gradually refilled.

The results of RELAP5 and TRAC Break Spectrum analysis reported in Reference 5 indicate that the smaller break size tends to enhance the differences in event timing. It should be noted that these calculations are limited to simulation of early cooldown phase. For the purpose of more realistic assessment of the boron dilution and relevant mixing processes the thermal-hydraulic conditions during the refill phase of small-break LOCAs and the effects of secondary side depressurization on primary inventory recovery and reestablishment of natural circulation should be further evaluated.

2.2 Natural Circulation Phenomena

Natural circulation is an essential decay heat removal mechanism during small-break LOCAs. There are three distinct modes of natural circulation within the RCS: single phase, two phase (liquid continuous) and reflux condensation. An important conclusion reached from the model tests for PWRs is that the occurrence of a given natural circulation mode is principally a function of primary system coolant mass inventory.[7]

Single-phase natural circulation provide heat redistribution from the core to the steam generators after the RCPs coast down. By integrating the loop momentum equation and expressing the density variation in terms of the volumetric thermal expansion coefficient, the steady state single phase natural circulation flow can be expressed as:[8]

$$W_{1\phi} = \left[\frac{2\beta g \rho_l^2 Q_o \Delta L}{C_p R_f}\right]^{\frac{1}{3}} \qquad (1)$$

Where Q_o is the core decay heat rate and ΔL is the elevation difference between the thermal centers of the core (the heat source) and the steam generator (the heat sink). The flow resistance parameter R_f is taken as that for forced flow in the system.[8] A comparison of the theoretical predictions using Equation 1 with the test results for Zion Unit 1[6] (Westinghouse four loops design, 3250 MWt) is shown in Figure 3.

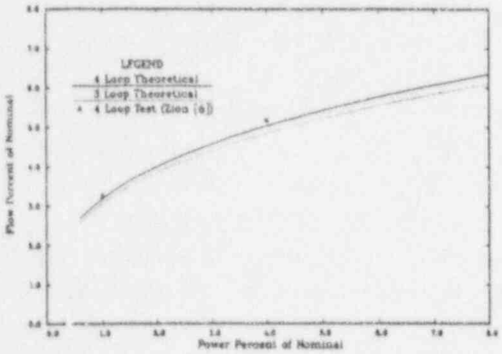


Figure 3 Comparison of single phase natural circulation calculations with test data for Zion.

The single-phase natural circulation occurs from 100% primary coolant inventory down to the onset of voiding in the upper plenum above the hot leg elevation.[7] Further depletion of coolant causes the upper plenum voiding to reach the elevation of hot leg. Initially steam produced by flashing in the reactor vessel will be condensed soon after it enters the inlet to the steam generator tubes, and natural circulation flow will be maintained and even increases because of the overall coolant density in the core, hot leg and upflow side of steam generators decreases.

Continued loss of reactor coolant inventory through the break causes the vessel upper plenum, hot leg and the steam generator primary side become predominantly vapor filled. With this condition the primary mode of natural circulation cooling is reflux condensation. Steam generated in the core rises through the hot legs to the steam generator tubes. The portion of steam that condenses on the upflow side of the U tubes may flow back to the vessel through the hot leg or is carried over to the downflow side. The carryover, with the remaining condensation in the downflow side drops into the pump suction piping (pump seal). The results of Semiscale natural circulation tests[9] shows that the reflux flow rate is almost 50% of the core vapor generation (see Figure 2.6). As it is pointed out in Reference 7, this ratio remained relatively constant, independent of core power, secondary side inventory and primary system inventory. The accumulating condensate mass flow into the loop seals under steady-state phase separating natural circulation, q_a, can be expressed by:

$$q_a = \frac{0.5 Q_o}{(h_{fg} + \Delta h)} \qquad (2)$$

Where h_{fg} and Δh are the latent heat and inlet subcooling respectively.

Utilizing the quasi-steady hypothesis and by solving analytically the loop momentum balance together with conservation of mass and energy, Duffey and Sursock[10] obtained expressions for the core flow rate as a function of inventory. Their model predictions has been shown to be in good agreement with the experimental data from the Semiscale[10] and FLECHT-SEASET[11] facilities. For a detail discussion of their simplifying assumptions and model formulation refer to Reference 10. Here we used the Duffey and Sursock model and developed a map of natural circulation flow rate (fraction of nominal) versus normalized mass inventory for 2% decay power level for different RCS pressure conditions as shown in Figure 4. This map can be used for estimating the natural circulation flow under increasing inventory (refill phase).

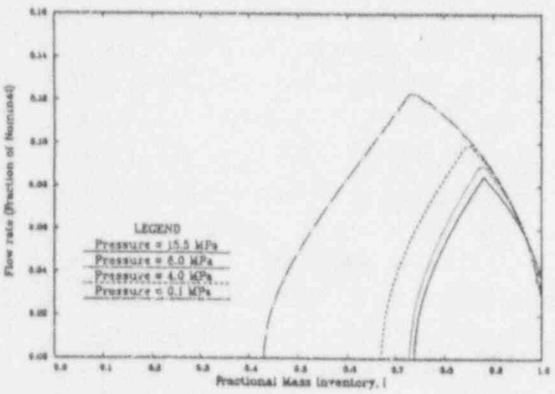


Figure 4 Calculated natural circulation flow rate as a function of fractional mass inventory and pressure (2% decay power).

2.3 Loop Seal Clearing

A typical pressurized water reactor has U-shaped crossover pipes which connect the steam generator outlet plenum to the reactor coolant pump (see Figure 2). During small-break LOCAs, steam passing through the steam generators may blow the water in the crossover legs out to either the break or vessel (loop seal clearing).

Experimental studies on loop seal clearing indicate that the loop seals are cleared only after the liquid level in the vertical leg below the steam generator outlet plenum reaches the top of the bottom horizontal section.[12-13] This seemingly simple phenomena should be clearly recognized for analysis of boron mixing prior to loop seal clearing.

3 MIXING PROCESSES AND PHENOMENA

In Section 2, the potential for accumulation of diluted water in the loop seals due to the reflux condensation mode of operation during certain small-break LOCAs was discussed. However, the buoyancy and turbulent mixing along the way from the loop seals to the core may sufficiently increase the boron concentration of the diluted stream to prevent a power excursion leading to fuel failure.

In this section, the quantitative aspects of different mixing mechanisms are first presented and a methodology for their integration into an overall prediction of dilution boundary is discussed. Bounding calculations for the concentration of boron in the coolant entering the core during the refill phase of a small break LOCA and the re-establishment of natural circulation flow in a Westinghouse 4-loop plant will be presented in Section 4.

3.1 Mixing in the Loop Seals

During reflux condensation the loop mean flow rate is virtually null. The safety injection of cold, highly borated water into a stagnant loop of a PWR leads to stratification accompanied by counter-current flows and recirculation. The ensuing flow regime was first established analytically by Theofanous and Nourbakhsh[14,15] as part of the work in support of NRC Pressurized Thermal Shock (PTS) study to predict the overcooling transients due to high pressure safety injection into a stagnant loop of a PWR. The physical situation may be described with the help of Figure 5. A "cold stream" originates with the safety injection buoyant jet at the point of injection, continues toward both ends of the cold leg, and decays away as the resulting buoyant jets fall into the downcomer and pump/loop-seal regions. A "hot stream" flows counter to this "cold stream" supplying the flow necessary for mixing (entrainment) at each location. This mixing is most intensive in certain locations identified as mixing regions (MRs).

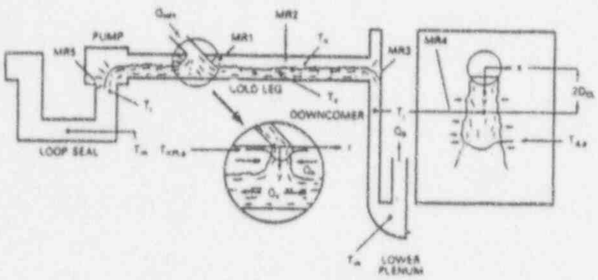


Figure 5 Conceptual definition of flow regime and regional mixing model.

MR1 indicates the mixing associated with the buoyant, nearly axisymmetric safety injection jet. MR3 and MR5 are regions where mixing occurs because of transions (jumps) from horizontal layers into falling jets. MR4 is the region where the downcomer (planar) buoyant jet finally decays. The cold streams have special significance because they induce a global recirculating flow pattern with flow rates significantly higher than the net flow through the system. This keeps a major portion of the system volume including the loop seals (vertical leg below the pump and bottom horizontal leg), the downcomer (excluding the region above the cold leg), and the lower plenum in a well mixed condition.

The quantitative aspects of this physical behavior may be found in the regional mixing model.[15,16] This model accounts for countercurrent flow limitations between the cold and hot streams at the cold leg/downcomer junction and incorporates plume mixing rates which are consistent with data from idealized single plume geometries. The regional mixing model and the associated computer program REMIX[17] has been successfully employed to the interpretation of all available thermal mixing experimental data obtained from the system simulation tests performed in support of the PTS study.[18]

A similar thermal stratification and mixing behavior may even exist in the presence of low loop mean flow. The criterion for the existence of thermal stratification in the presence of loop flow will be discussed in Section 3.2. In the presence of thermal stratification and effective natural recirculating flows, the dilution transient can be represented by a simple global boron mass conservation equation:

$$\rho V \frac{dC_m}{dt} = q_{SI}(C_{SI} - C_m) + q_L(C_L - C_m) \quad (3)$$

Where ρ is the density (the effect of density variation is neglected); V is the system volume; C_m, C_{SI} and C_L are boron concentrations of flow entering the core, safety injection and loop flow (entering the bottom horizontal leg of loop seal), respectively; and q_{SI} and q_L are the safety injection and loop flows, respectively. It should be noted that the volume V includes the cold leg, pump, lower plenum, downcomer (excluding the portion above the cold leg) and the vertical leg below the pump and bottom horizontal leg of the loop seal. The downcomer and lower plenum volumes should be partitioned equally among the available loops.

Equation 3 can be integrated analytically to:

$$C_m = \frac{C_{SI} + RC_L}{1+R} + \left[C_o - \frac{C_{SI} + RC_L}{1+R}\right] e^{\frac{(1+R)t}{\tau}} \quad (4)$$

where

$$\tau = \frac{\rho V}{q_{SI}} \quad (5)$$

and

$$R = \frac{q_L}{q_{SI}} \quad (6)$$

Assuming that initially, the system is filled with borated water with a boron concentration of 1500 PPM, the time variation of boron concentration, Cm, due to loop flow of unborated water (C_{LS} = 0) for q_{SI} = 7 Kg/sec, C_{SI} = 2200 PPM and different values of R is illustrated in Figure 6. Neglecting the variation of condensate level in the vertical downflow leg, the flow of condensate entering the bottom horizontal leg, q_L, can be estimated from Equation 2. For example at a pressure of 8.MPa and with the assumption that three steam generators stay active, q_L = 7 Kg/sec and R ≈ 1. With a flow ratio of R = 1, the boron concentration would be more than 1100 PPM.

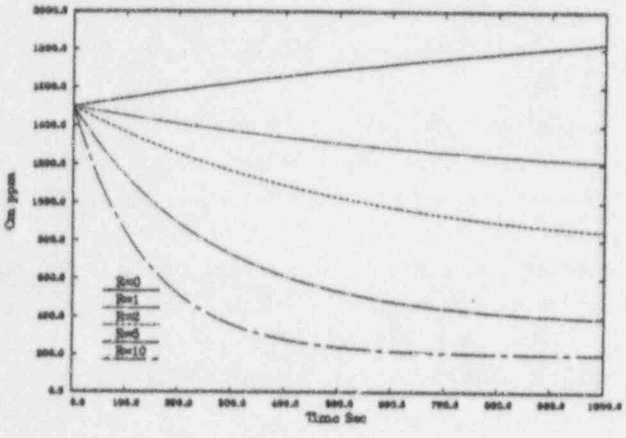


Figure 6 Dilution transient under stratified and recirculating flow regime.

3.2 Mixing at the Safety Injection Point

For a well mixed condition (see Figure 7) there must be sufficient loop flow not only to break up the safety injection plume (jet) but also to produce stable flow into the downcomer. Nourbakhsh and Theofanous[19] used the boundary of stability and developed a criterion for the existence of perfect mixing in the presence of loop flow, their stratification/mixing boundary can be expressed by:

$$Fr_{SI,CL} = \left(1 + \frac{Q_L}{Q_{SI}}\right)^{-7/5} \quad (7)$$

where Q_{SI} and Q_L are the volumetric flow rates of the safety injection and the loop, respectively. The Froude number, $Fr_{SI,CL}$ is defined as:

$$Fr_{SI,CL} = \frac{Q_{SI}}{A_{CL}} \left\{ g D_{CL} \frac{(\rho_{SI} - \rho_L)}{\rho_{SI}} \right\}^{-1/2} \quad (8)$$

where A_{CL} and D_{CL} are the flow area and the diameter of cold leg respectively.

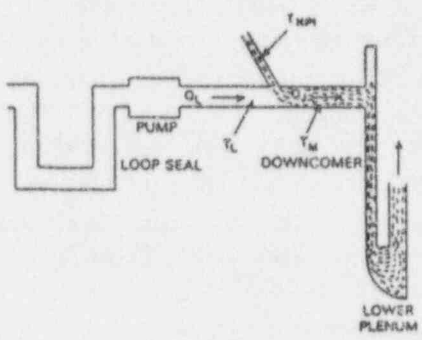


Figure 7 Conceptual representation of the well-mixed condition[19]

This stratification criterion should be considered as providing a high estimate of flow ratio, R, necessary for ignoring stratification. For perfect mixing, the concentration of diluted flow stream after mixing with the safety injection flow, C_{pm} can be easily quantified by the boron mass balance at the mixing point.

$$C_{pm} = \frac{C_{SI} + RC_L}{1+R} \tag{9}$$

Typically, natural circulation flows are in the 110 to 250 Kg/sec range. For a RCS pressure of 4.2 MPa, the safety injection flow is ~ 10 Kg/sec. In terms of stratification criterion parameters, these values correspond to $Fr_{SI,CL} \approx 0.02$ and $R = 15$, indicating perfect mixing except for the lower range of natural circulation flow.

3.3 Mixing in the Downcomer

A highly complicated three dimensional mixing pattern occurs at the cold leg-downcomer junction.[15] This contribution to mixing is conservatively neglected and the dilute stream exiting the cold-leg is assumed to form smoothly into a planar plume within the downcomer. Under low loop flow condition, the diluted stream entering the downcomer would be colder than the downcomer coolant due to mixing with the safety injection. The resulting positively buoyant planar jet decay rapidly, enhancing the mixing and global flow recirculation. However, in the presence of relatively high natural circulation loop flow, the temperature of condensate, even after the mixing with the safety injection flow would be higher than the downcomer temperature and thus the inlet flow into the downcomer constitute a negatively buoyant jet (Inverted Fountain). A schematic of both positive and negative buoyant jets is illustrated in Figure 8.

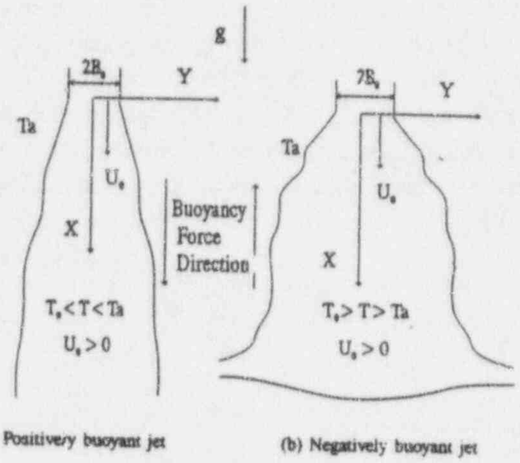


Figure 8 Schematic of a positively and a negatively buoyant jet.

Except for limited data on maximum penetration distance[20], there have been no experimental or analytical studies on the behavior of negatively buoyant planar jets reported in the open literature. In order to be able to quantify the mixing of a negatively buoyant planar jet of the diluted water with the highly borated downcomer ambient, an extensive analytical study of negative buoyant jets was performed as a part of the present work.[21] The jet model of Chen and Rodi[22] was adopted for this purpose. The model utilizes the standard equations for natural convection boundary layer type flows with a vertically oriented buoyance force and a $K\text{-}\epsilon\text{-}\overline{T}^2$ differential turbulence model to evaluate the transport terms in the equations. With the choice of appropriate scales, these equations may be put in dimensionless form such that only one main parameter, the Froude number, appears.

The integration was carried out using the Patankar-Spalding method.[23] In order to achieve high computational efficiency, this method invokes a coordinate transformation, which utilizes a normalized Von Mises variable; and thus instead of the y coordinate, a nondimensional stream function is used in the transverse coordinate.

Results for the Froude number of interest here (Fr = 1.5) are presented in Figures 9 and 10. It should be noted that the nondimensional axial and transverse direction X^*, Y^*, Froude number, Fr, nondimensional temperature (or concentration), T^*, and nondimensional velocity U^* are defined as follows:

$$X^* = \frac{x}{2B_o} \tag{10}$$

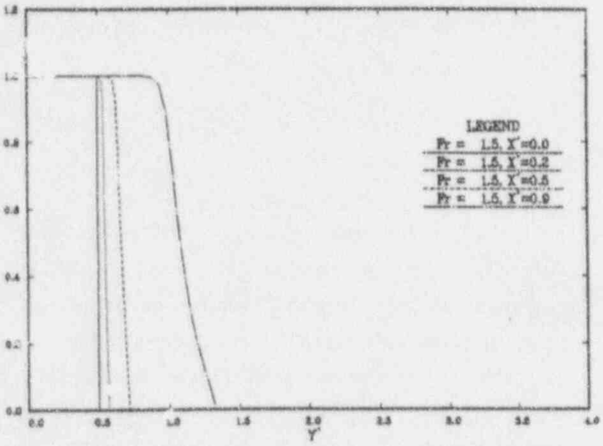


Figure 9 Calculated results of temperature (or concentration) profiles for a negatively buoyant planar jet (Fr = 1.5)

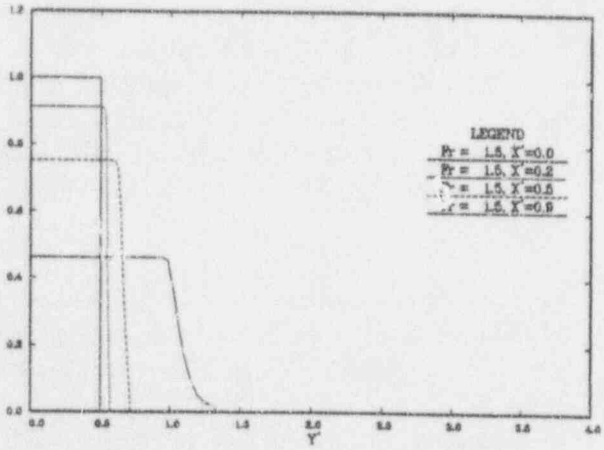


Figure 10 Calculated results of velocity profiles for a negatively buoyant planar jet (Fr = 1.5)

$$Y* = \frac{Y}{2B_o} \tag{11}$$

$$T* = \frac{T-T_a}{T_o-T_a} = \frac{C-C_a}{C_o-C_a} \tag{12}$$

$$U* = \frac{U}{U_o} \tag{13}$$

$$Fr = \frac{U_o}{\sqrt{2B_o g(\rho_a-\rho_o)/\rho_o}} \tag{14}$$

The results of the turbulent jet model illustrate that at low Froude number, the negatively buoyant planar jets spread rapidly in the lateral dimension with much lower entrainment or mixing as compared to positively buoyant planar jets. For example, a negatively buoyant planar jet with a Froude number of 1.5 decelerates to less than 50% of its initial velocity, without any significant entrainment or mixing, within less than one initial width of the jet ($X^* < 1$). In a negatively buoyant jet, due to buoyancy force which acts against the flow direction, the flow penetrates to a finite distance in the ambient environment before reversal occurs. It should be noted that the present Parabolic turbulent jet model neglects the effect of return flow. Furthermore, the validity of boundary layer assumptions is questionable near the stagnation point where the axial velocity, being sufficiently reduced, approaches zero and significant lateral spreading of the jet occurs.

The general adequacy of the turbulent jet model utilized here has been demonstrated by Chen, Rodi and co-workers[22,24,25] for positively buoyant jets. To ensure the applicability of turbulence model for negatively buoyant jets the model was used to predict the maximum height of negatively buoyant round (axisymmetric) jets.[21] The maximum height was defined as the point where the centerline velocity decays to 1% of the discharging velocity. The result was found to be in excellent agreement with Turner's correlation,[26] deduced from his experimental data of the penetration height of salt water injected upward into fresh water.

The turbulent jet model was also utilized to predict the maximum penetration distance for negatively buoyant planar jets.[27] If the source is small compared with the maximum penetration distance, the flow will depend only on the buoyancy flux, F_o, and momentum flux, M_o, at the jet source. In this case, the flow will not depend explicitly on the volume flux, Q_o. Following an approach similar to the one used by Turner for the case of a circular fountain,[26] the maximum penetration distance of a negatively buoyant planar jet, H_{max}, can be defined by the dimensional consistency requirement as:

$$H_{max} = \text{constant} \times \left(\frac{M_o}{\rho_o}\right)\left(\frac{F_o}{\rho_o}\right)^{-2/3} \quad (15)$$

where

$$M_o = 2B_o \rho_o U_o^2 \quad (16)$$

$$F_o = 2B_o\, \rho_o\, U_o\, g\, (\rho_a - \rho_o)\,/\rho_a \quad (17)$$

Combined with the definition of densimetric Froude number, (Eq. 14), the maximum penetration distance (Eq. 15) can be expressed by:

$$\frac{H_{max}}{2B_o} = \text{constant} \times F_r^{4/3} \quad (18)$$

The proportionality constant evaluated by the turbulent jet model predictions is 2.42 as shown in Figure 11. Due to computational difficulty, it was not possible to predict the maximum penetration distance for low Froude number jets ($Fr<3$). Assuming that at low Froude number the flow depends on momentum flux and volume flux only, based on the dimensional consistency requirement, the maximum penetration distance, $H_{max}/2Bo$, should be a constant. This is also supported by the experimental data reported by Goldman and Jaluria[19] which indicate a finite value of penetration distance as Froude number decreases to a very low value. Thus, the maximum penetration distance for $Fr>2$ can be correlated by:

$$\frac{H_{max}}{2Bo} = 2.42\ Fr^{\frac{4}{3}}\ for\ Fr > 2 \quad (19)$$

A comparison of the present correlation with the data reported by Goldman and Jaluria is also presented in Figure 11.

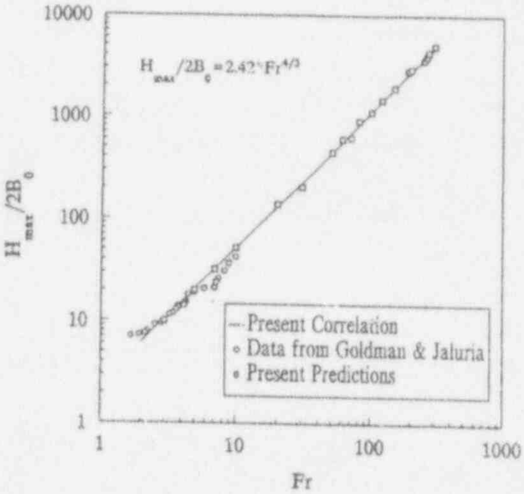


Figure 11 Comparison of predicted maximum penetration distance of negatively buoyant vertical planar jets with experimental data.

Assuming that the ambient to the negatively buoyant planar jet in the downcomer behaves as though it is well mixed, the global boron mass conservation equation can be expressed as:

$$\rho V_a \frac{dC_a}{dt} = q_{ent}(C_{mj} - C_a) \qquad (20)$$

$$C_{mj} = \frac{C_o + \frac{q_{ent}}{q_o} C_a}{1 + \frac{q_{ent}}{q_o}} \qquad (21)$$

Where ρ is the density (effect of density variation is neglected); C_o, C_a and C_{mj} are boron concentrations of flow entering the downcomer, the ambient, and mean jet flow entering the lower plenum, respectively; V_a is the volume of ambient, and q_o and q_{ent} are the inlet flow to downcomer and entrainment flow into planar jet, respectively. It should be noted in the case of symmetric flow of diluted water from different loops, the low Froude number negatively buoyant jets entering the downcomer grow rapidly in the lateral direction and thus would occupy the whole downcomer circumference before reaching to the lower plenum. In this case, the volume of ambient would be reduced accordingly (see Section 4).

Equations 20 and 21 can be integrated analytically to:

$$\frac{C_a - C_o}{C_a^o - C_o} = e^{-\frac{\alpha}{1+\alpha}\frac{q_o}{\rho V_a}t} \qquad (22)$$

$$\frac{C_{mj} - C_o}{C_a^o - C_o} = \frac{\alpha}{1+\alpha} e^{-\frac{\alpha}{1+\alpha}\frac{q_o}{\rho V_a}t} \qquad (23)$$

where

$$\alpha = \frac{q_{ent}}{q_o} \qquad (24)$$

The numerical values of α can be obtained from the results of jet model.

3.4 Mixing In The Lower Plenum

The diluted stream of water leaving the downcomer will experience some mixing with the highly borated water of the lower plenum before entering the core.

In the presence of thermal stratification (very low loop flow), due to entrainment, the positively buoyant planar wall jet entering the lower plenum carries a flow which is at least one order of magnitude higher than the HPI flow. Thus, the highly borated water in the lower plenum is drawn continuously to the downcomer and cold leg resulting in a very intensive mixing and recirculation in the lower plenum. Indeed the results of thermal mixing experiments related to pressurized thermal shock[18] (under stagnated flow condition) indicate no thermal stratification in the lower plenum (well mixed lower plenum).

Under the relatively high natural circulation loop flow (even in the presence of stratification), the loop flow accommodates a significant portion of entrainment and thus there may not be significant recirculation (if any) from the lower plenum back to downcomer. However, the negatively buoyant wall jet of diluted water entering the lower plenum will penetrate to some finite depth before it reaches to a stagnation point and then reverses direction upward toward the core region. The highly borated ambient water in the lower plenum will be entrained into this flow, resulting in higher boron concentration of flow entering the core compared with that entering the lower plenum. The detailed quantification of mixing in lower plenum is beyond the scope of the present study. Some bounding calculations to show the impact of lower plenum mixing are presented in Section 4.

4 BOUNDING ANALYSES

Many thermal hydraulic aspects of boron dilution, except the mixing effects, can be analyzed by using system codes, such as TRAC and RELAP. The mixing processes underway from the loop seal to core involve multidimensional one-phase flow effects which typically are not modelled in system codes. Furthermore, these codes exhibit far too much numerical diffusion to be useful for tracking of a relatively sharp concentration gradient around the system.[1] Simulation of dilution transients using one of the system codes to provide the thermal hydraulic conditions needed for both the mixing analysis and the reactor physics calculations is beyond the scope of the present study.

The boron dilution necessary to cause fuel damage depends on many factors including the initial shutdown margin, the doppler feedback, the delayed neutron fraction, the neutron lifetime and the speed at which the slug of diluted water moves through the core. The consequence analysis to predict the effect of dilution on fuel integrity is beyond the scope of the present study. However, it should be noted that the results of neutronics calculations,[28] based on an approximate synthesis method, obtained within the context of externally caused rapid boron dilution (with an insurge slug velocity corresponding to 13% of rated flow) indicate that a slug with a concentration of 750 ppm entering the core region with an initial 1500 ppm concentration, could result in an excursion that breaches reactivity insertion accident (RIA) criteria.

In this section, bounding calculations for boron concentration of coolant entering the core due to subsequent change in flow conditions such as loop seal clearing or re-establishment of natural circulation flow in a typical Westinghouse design 4-loop plant (RESAR-3S) are presented.

4.1 Boron Dilution Due to Loop Seal Clearing

Loop seal clearing has been suggested as a potential mechanism for driving an accumulated slug of diluted water from the loop seals into the core.[3] The loop seals are cleared only after the liquid level in the vertical leg below the steam generator outlet plenum reaches the top of the bottom horizontal section (see Section 2.3). During this period of gradual reduction of liquid level in the vertical leg, the loop flow entering the bottom horizontal leg of the loop seals is relatively low. As discussed in Section 3 (under low loop flow conditions) the safety injection of cold, highly borated water into the cold leg leads to stratification accompanied by counter-current flows and recirculation. For example at a pressure of 8.MPa, and with the assumption that three steam generators stay active, the safety injection flow, q_{SI}, is 7 Kg/sec. The flow of condensate entering the bottom horizontal leg, q_L, based on a condensation rate (Eq. 2) and the TRAC results of loop seal level change for a 3 in. break reported in Reference 5, is estimated to be $\approx$ 11.9 Kg/s. Under these conditions, $Fr_{SI,CL}$ = 0.013 and R $\approx$ 1.7, indicating flow stratification. Thus, the resulting boron concentration of flow entering the core, Cm, can be estimated by using Equation (4) (see also Figure 6). Assuming that initially the system is filled with borated water with a boron concentration of 1500 PPM, the boron concentration after 350 seconds (based on time duration of level reduction before loop seal clearing reported in Reference 5) is more than 1200 ppm.

4.2 Boron Dilution Due to Reestablishment of Natural Circulation Flow

The re-establishment of natural circulation flow may occur during the refill phase of small break LOCAs (SBLOCAs) as long as the secondary heat sink is available. The magnitude and timing of the natural circulation flow depend on plant geometry, break size and location, ECCS capacities, equipment failure criteria and operational actions, all of which differ considerably in various designs. In the absence of detailed system code simulation results during the refill phase of small break LOCAs, bounding estimates of the needed thermal-hydraulic conditions fo mixing calculations was used to predict the dilution boundary.

If the RCS refill and re-establishment of natural circulation proceed at low pressure (a characteristic of relatively large SBLOCAs) high flow of cold, highly borated water injected into the cold leg, via accumulators or low pressure safety injection system, mixes with the natural circulation flow of unborated water. This leads to a significant increase in boron concentration of the resulting flow before entering the core region. For example for a typical Westinghouse- designed 4-loop plant, the low pressure safety injection flow is on the order of 115 Kg/sec per loop. The natural circulation flow, estimated by conservatively assuming 2% core decay power is about 4% of the nominal flow or 175 Kg/s per loop. Neglecting the potential for thermal stratification and conservatively assuming perfect mixing, the boron concentration of the resulting flow entering the downcomer estimated by using Equation (9) is 872 ppm. Even without considering any mixing in the downcomer and lower plenum, this level of boron concentration does not result in a power excursion leading to fuel failure.

For the present bounding analyses, it was also assumed that the reestablishment of natural circulation occurs at a RCS pressure higher than accumulator injection setpoint (a characteristic of relatively smaller SBLOCAs). Assuming a RCS pressure of 4.8 MPa ($\approx$ 700 Psia), decay power of 2% and with the assumption that three steam generators stay active, the estimated two phase natural circulation flow under increasing inventory (based on Figure 4, assuming 36 Kg/sec net refill rate) is 112 Kg/sec. The boron concentration of flow after mixing with the HPI injection of 12 Kg/sec (perfect mixing condition) is 213 PPM. The resulting flow of 124 Kg/sec into downcomer was assumed to form a planar jet with an initial Froude number of $\approx$ 1.5. Using the mixing models presented in Section 3 with $V_s = 0.3m^3$ and $\alpha = 0.08$, the resulting transient boron concentration entering the core was calculated for total condensate volume of $4m^3$ and $10m^3$. Calculations were performed under two limiting conditions of mixing in lower plenum as shown in Figures 12 and 13. Sensitivity calculations were also performed with a bounding estimate of single phase natural circulation flow of 225 Kg/sec per loop as shown in Figure 14.

5 SUMMARY AND CONCLUSIONS

A scoping study of boron dilution and mixing phenomena during small break LOCAs in pressurized water reactors was performed. The mixing processes associated with a slow moving stream of diluted water through the loop seals to the core were examined. The quantitative aspects of different mixing mechanisms and a simplified, yet physically based, methodology for their integration into an overall prediction of dilution boundary were presented. Bounding case analyses for boron concentration of coolant entering the core due to loop seal clearing or re-establishment of natural circulation flow in a typical Westinghouse-designed, 4-loop plant were also presented.

During reflux condensation, the loop mean flow rate is virtually null. Based on the results of thermal mixing studies related to pressurized thermal shock issue, the safety injection of cold and highly borated water into a stagnant loop leads

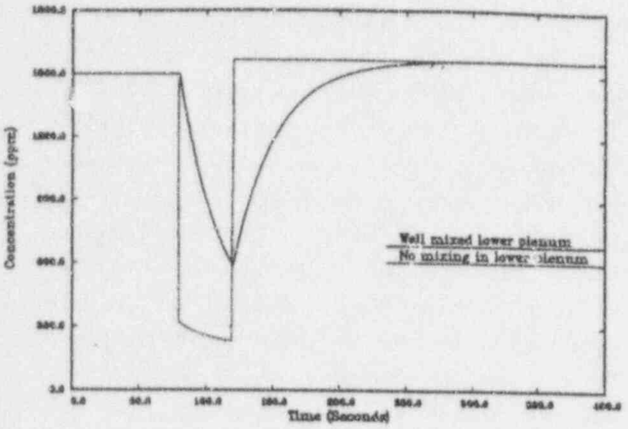


Figure 12 Transient boron concentration entering the core (condensate volume = 4m³, natural circulation flow = 112 Kg/sec)

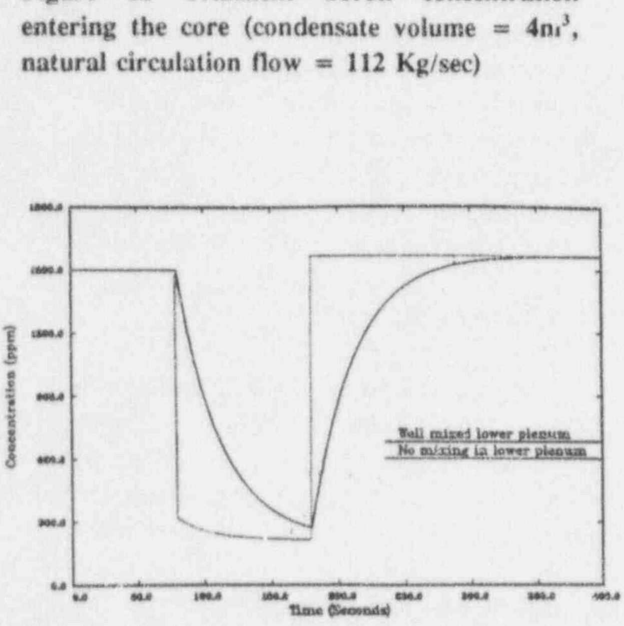


Figure 13 Transient boron concentration entering the core (condensate volume 10m³, natural circulation flow 112 Kg/sec)

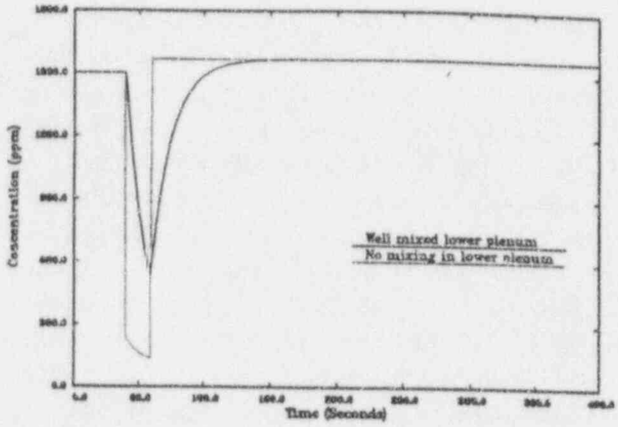


Figure 14 Transient boron concentration entering the core (condensate volume = 4m³, natural circulation flow = 225 Kg/sec)

to stratification accompanied by counter-current flows and a global recirculating flow pattern with flow rates significantly higher than the net flow through the system. This keeps a major portion of the system volume including the loop seals (vertical leg below the pump and bottom horizontal leg), the downcomer (excluding the region above the cold leg), and the lower plenum in a well-mixed condition. Assuming that a similar thermal stratification and mixing behavior may even exist in the presence of low loop mean flow, the boron concentration of flow entering the core was obtained analytically by integrating the global boron mass conservation equation.

Under low loop flow conditions, the diluted stream entering the downcomer would be colder than the downcomer coolant due to mixing with safety injection. The resulting positively buoyant planar jet decays rapidly, enhancing the mixing and global flow recirculation. However, in the presence of relatively high natural circulation loop flow, the temperature of condensate, even after mixing with the safety injection flow, would be higher than the downcomer temperature and thus the inlet flow into the downcomer constitutes a negatively buoyant planar jet (inverted fountain). In order to be able to quantify the mixing of a negatively buoyant planar jet of the diluted water with the highly borated downcomer ambient, an extensive analytical study of negatively buoyant planar jets was performed as a part of the present work. The differential turbulence model of Chen and Rodi was adopted for this purpose. Using dimensional analysis and the results of the turbulent jet model, a correlation for the maximum penetration distance of a negatively buoyant planar jet, as a function of densimetric Froude number, was also obtained as a part of this study.

Experimental studies on loop seal clearing indicate that the loop seals are cleared only after the liquid level in the vertical leg below the steam generator outlet plenum reaches the top of the bottom horizontal section. During this period of gradual reduction of liquid level in the vertical leg, the safety injection of cold and highly borated water into the cold leg leads to stratification accompanied by counter-current flow and recirculation. An illustrative prediction for a typical Westinghouse-designed, 4-loop plant indicates that the boron concentration of flow entering the core does not fall below 1200 PPM when the initial boron concentration in the vessel is 1500 PPM.

If the RCS refill and re-establishment of natural circulation flow proceeds at low pressure (a characteristic of relatively large SBLOCAs), the loop flow of unborated water mixes with the highly borated water injected into the cold leg via accumulators and the low pressure safety injection system. This leads to a significant increase in boron concentration of the resulting flow before it enters the core region.

For the present scoping analyses, it was also assumed that the re-establishment of natural circulation flow occurs at a RCS pressure higher than the accumulator injection setpoint (a characteristic of relatively smaller SBLOCAs). For the bounding cases considered, the boron concentration of the loop flow after mixing with the high-pressure injection (HPI) is 213 PPM. The resulting low Froude number negatively buoyant jets entering the downcomer (assuming symmetric flow of diluted water from different loops) grow rapidly in the lateral direction (without significant mixing) and will occupy the entire downcomer circumference before reaching the lower plenum. Sensitivity calculations indicate the importance of quantification of mixing in the lower plenum for a more realistic prediction of boron concentration entering the core region.

A more realistic assessment of the boron dilution and relevant mixing processes requires further evaluation of the thermal-hydraulic conditions during the refill phase of small break LOCAs, and the effects of secondary side depressurization on primary inventory recovery and reestablishment of natural circulation flow. Detailed quantification of mixing in the lower plenum is also desirable.

ACKNOWLEDGEMENTS

The authors wish to extend their appreciation to Dr. D. Ebert of the U.S. Nuclear Regulatory Commission and Professor A. Baretta of Pennsylvania State University for their support and encouragement.

NOMENCLATURE

Latin Letters

A	area
B	initial half width of a planar jet
C	concentration
D	diameter
F_o	buoyancy flux at the jet source
Fr	Froude number
g	gravitational acceleration
H	height
M_o	momentum flux at the jet source
Q	volumetric flow rate

q	mass flow rate
R	flow ratio (see Eq. 4)
T	temperature
t	time
U	velocity
V	volume
X	coordinate in axial direction
Y	coordinate in transverse direction

Greek Letters

α	nondimensional entrainment flow (see Eq. 22)
ρ	density
τ	mixing time constant under no loop flow condition (see Eq. 3)

Subscripts

o	initial
α	ambient
CL	cold leg
ent	entrainment
L	loop
m	mixed mean
max	maximum
SI	safety injection

REFERENCES

1. S. Jacobson, "Risk Evaluation of Local Dilution Transients in a Pressurized Water Reactor," Linköping Studies in Science and Technology, Dissertation No. 275.

2. K. Haule and J. Hyvärinen, "Potential Reactivity Problems Caused by Boron Separation/Dilution During Partial Loss of Inventory Transients and Accidents in PWR's," STUK/YTO Memo, Dec. 1991.

3. J. Hyvärinen, "An Inherent Boron Dilution Mechanism in Pressurized Water Reactors," The 5th International Topical Meeting on Reactor Thermal Hydraulics, NURETH-5, Salt Lake City, USA, 1992.

4. W. E. Burchill, "Physical Phenomena of a Small-Break Loss-of-Coolant Accident in a PWR," Nuclear Safety, Vol. 23, No. 5, 1982.

5. C. D. Fletcher and C. M. Kullberg, "Break Spectrum Analysis for Small Break Loss of Coolant Accidents in a RESAR-3S Plant" NUREG/CR-4384, INEL, March 1986.

6. U. S. Nuclear Regulatory Commission, "Generic Evaluation of Feedwater Transients and Small Break Loss-of-Coolant Accidents in Westinghouse-Designed Operating Plants" NUREG-0611, January 1980.

7. D. J. Shimeck and G. W. Johnson, "Natural Circulation Cooling in a Pressurized Water Reactor Geometry Under Accident Conditions," Nuclear Science and Engineering, Vol. 88, 311-320, 1984.

8. Y. Zvirin, P.R. Jeuck, C. W. Sullivan, and R. B. Duffey, "Experimental and Analytical Investigation of a Natural Circulation System with Parallel Loops," Trans. ASME. J. of Heat Transfer Vol. 103, 645-652, 1981.

9. G. G. Lomis and K. Soda, "Results of the Semiscale Mod-2A Natural Circulation Experiments" NRC Report, INEL, NUREG/CR-2335, 1982.

10. R. B. Duffey and J. P. Sursock, "Natural Circulation Phenomena Relevant to Small Breaks and Transients," Nuclear Engineering and Design, Vol. 102, 115-128, 1987.

11. L. E. Hochreiter, et.al., "PWR Flecht Seaset Systems - Effects Natural Circulation and Reflux Condensation Data

Evaluation and Analysis Report," EPRI/NRC/Westinghouse Report, NP-3497, 1985.

12. R. J. Skwarek, "Experimental Evaluation of PWR Loop Seal Behavior During Small LOCA's" Proceedings of ANS Specialist Meeting on Small Break Loss-of-Coolant Analyses in LWR's, Monterey, CA, August 25-27, 1981.

13. N. Lee, "Discussion on Loop Seal Behavior During Cold Leg Small Break LOCAs of a PWR," Nuclear Engineering and Design, Vol. 99, 453-458, 1987.

14. T. G. Theofanous and H. P. Nourbakhsh, "PWR Downcomer Fluid Temperature Transients Due to High Pressure Injection at Stagnated Loop Flow," Proc. Joint NRC/ANS Meeting on Basic Thermal Hydraulic Mechanisms in LWR Analysis, September 14, 15, 1982, Bethesda, Maryland, NUREG/CP-0043, 583-613, 1983.

15. H. P. Nourbakhsh and T. G. Theofanous, "Decay of Buoyancy-Driven Stratified Layers with Application to Pressurized Thermal Shock, Part I: The Regional Mixing Model," Nuclear Eng. & Design, (in press).

16. T. G. Theofanous, H. P. Nourbakhsh, P. Gherson and K. Iyer, "Decay of Buoyancy-Driven Stratified Layers with Application to Pressurized Thermal Shock," NUREG/CR-3700, U.S. Nuclear Regulatory Commission, May 1984.

17. K. Iyer, H. P. Nourbakhsh and T. G. Theofanous, "REMIX: A Computer Program for Temperature Transients Due to High Pressure Injection After Interruption of Natural Circulation," NUREG/CR-3701, May 1986.

18. T. G. Theofanous and H. Yan, "A Unified Interpretation of One-Fifth to Full-Scale Thermal Mixing Experiments Related to Pressurized Thermal Shock," NUREG/CR-5677, 1991.

19. H. P. Nourbakhsh and T. G. Theofanous, "A Criterion for Predicting Thermal Stratification Due to High-Pressure Injection in a Circulating Reactor Loop," Nuclear Science & Engineering, Vol. 94, 77-79, September 1986.

20. D. Goldman and Y. Jaluria, "Effect of Opposing Buoyancy on the Flow in Free and Wall Jets," J. Fluid Mech., Vol. 166, pp. 41-56, 1986.

21. H. P. Nourbakhsh and Z. Cheng, "Potential for Boron Dilution During Small Break LOCAs in PWRs (A Scoping Analysis)," Technical Report W-6156, Brookhaven National Laboratory, 1994.

22. C. J. Chen and W. Rodi, "A Mathematical Model for Stratified Turbulent Flows and Its Application to Buoyant Jets" XVIth Congress, International Association for Hydraulic Research, San Paulo, 1975.

23. S. V. Patankar and D. B. Spalding, "A Finite Difference Procedure for Solving the Equations of the Two Dimensional Boundary Layer," Int. J. of Heat & Mass Transfer, Vol. 10, 1389-1411, 1968.

24. C. J. Chen and C. H. Chen, "On Prediction and Unified Correlation for Decay of Vertical Buoyant Jets," Journal of Heat Transfer, Vol. 102, 532-537, 1979.

25. C. J. Chen and C.P Nikitopoulos, "On the Near Field Characteristics of the Axisymmetric Turbulent Buoyant Jets in a

Uniform Environment," International Journal Heat & Mass Transfer, Vol. 22, 245-255, 1979.

26. J. S. Turner, "Jets and Plumes with Negative and Reversing Buoyancy," J. Fluid Mech., 26, 779-792, 1966.

27. H. P. Nourbakhsh and Z. Cheng, "Mixing Phenomena of Interest to Boron Dilution During Small Break LOCAs in PWRs," Proceeding of 7th International Meeting on Nuclear Reactor Thermal Hydraulics, NURETH-7, Saratoga Springs, NY, 1995.

28. D. J. Diamond, P. Kohut, H. Nourbakhsh, K. Valtonen, and P. Secker, "Probability and Consequences of Rapid Boron Dilution in a PWR," NUREG/CR-5819, BNL-NUREG-52313, 1992.

Multi-Dimensional Modeling of Boron Dilution Transients in the UMCP 2x4 Facility

J. Floyd, J. Green, K. Almenas and M. di Marzo*
Nuclear & Materials Engineering Department
*Mechanical Engineering Department
University of Maryland
College Park, MD 20742

Abstract

An experimental program at the University of Maryland College Park (UMCP) is investigating the generation, transport and mixing of boron-dilute volumes in a scaled model of a B&W lowered loop nuclear system. In conjunction with the experimental project, experience is being developed in the modeling of boron-dilution transients using multi-dimensional CFD codes. Scaling considerations necessitated a geometric difference between the model (UMCP) and prototype (TMI) downcomer gap. The effect of this difference upon the mixing of a boron-dilute volume prior to its reaching the core region is investigated using two and three dimensional models of the UMCP cold leg and downcomer. Computational models have been developed using both the Fluent and COMMIX-1C codes. Initial results show that for the UMCP facility geometry, differences between a widening of the downcomer gap on the core barrel side or the reactor pressure vessel side are minor.

Introduction

An experimental program is in progress at the University of Maryland College Park (UMCP) to investigate the mixing of boron depleted water volumes in a scaled model (linear scale ~1:4) of a Babcock & Wilcox lowered loop PWR. The research program and facility are described in a companion paper[1]. The research program deals with a broad spectrum of scenarios which investigate both the generation of boron free volumes and their subsequent transportation to the core region. The code simulations presented in this paper deal with one of these scenarios. It is assumed that a boron-dilute volume has been accumulated in the steam generator and cold leg loop seals and that a reactor cooling pump has been re-started. The primary interest is now focused on the degree of mixing which takes place before the boron free water reaches the lower core plane [2,3,4,5]. In such a scenario the diluted volume will be moved quickly along the cold leg piping and encounter its first directional change as it enters the annular downcomer. At this point flow in several directions is possible, in addition to the lowest resistance and thus predominant flow path downward into the lower RPV plenum, the plume can spread tangentially displacing water into the other cold legs. Because of large changes in flow geometry and direction, considerable turbulence and mixing is expected in this segment of the flow path. Tinoco[6] conducted related experiments for a 1:5 scaled volume of the Ringhals 3-loop Westinghouse plant. Conductivity probes placed preferentially at the entrance plane to the core revealed a strongly non-uniform concentration of the test solution.

Scaling goals for the UMCP 2x4 facility[1] specified the conservation of component volume ratios and flow path thermal-hydraulic resistances. These goals have been achieved. The modifications required to achieve it highlighted a scaling issue which so far has not been considered in the scaling literature. Namely: scaling criteria which are expressed in terms of volume ratios, hydraulic diameters, thermal-hydraulic resistance coefficients or other indexes of this nature are all inherently one dimensional. They cannot capture two-or three dimensional flow path aspects. Furthermore, since a number of two or three dimensional flow path geometries can be characterized by the same one-dimensional indexes, they are not unique. An example of such alternate flow geometries is shown in Figure 1. On the right hand side a cross section through the cold leg and downcomer of a B&W system is shown. As seen, a short distance below the cold leg entrance the downcomer becomes wider. As shown on the left hand side, in the model, this flow path feature is reproduced by reducing the lower core barrel diameter (this modification was necessary in order to match volumetric flow path component ratios). Thus in terms of linear indexes, like expansion ratios, the flow path characteristics can and have been matched; however, as shown in Figure 1, this does not make the two flow path's identical. In terms of the cold leg entrance, the off-set leading to the wider flow channel is located at opposing sides of the downcomer. In the prototype the widening is produced by a thinner RPV wall below the level at which the cold legs join the vessel and is thus on the RPV side. In the test facility the reduction of core region volume produced a widening on the core barrel side of the downcomer.

For many flow phenomena, such a difference has an either small or a completely negligible effect; however, it is not obvious that this is the case for mixing phenomena. An analytical investigation of such effects cannot use the classical one dimensional scaling methodology, and it must depend on multi-dimensional approaches. Fortunately, recent advances and cost reductions of computational hardware has made it possible to employ modern CFD techniques in a University environment. The Nuclear Engineering Program at the University of Maryland posses or has access to several such codes. The flow/mixing problem outlined above provides a realistic test case for a comparative analysis. This preliminary report thus has two purposes: one, to present initial results from these studies and two, to asses the suitability of CFD codes for such an application. The codes used in this assessment include Fluent[7], a commercial code, and COMMIX-1C[8], a code developed with NRC support.

Code Use Issues

Multidimensional CFD type codes have been under development for several decades. In the past they have helped in the resolution of the pressurized thermal stress (PTS) issue (Hassan[9], Theofanous[10,11]), and quite recently they have been applied to boron mixing problems (Alverez[12,13]). Nevertheless, their application in reactor safety related thermal hydraulic problems is not yet widespread. Therefore, it is worthwhile to report some of the experience accumulated during the course of this study. In a paper of this nature it is not possible to present a comprehensive inter-code comparison; thus, only the main observations which are likely to be useful to subsequent code users will be noted.

The two codes used in this analysis differ in a many respects, a number of the main differences are summarized in Tables I and II.

From the user point of view, the first distinction noted is the quite different approach to the 'input/output' issue. In this respect the COMMIX code falls into the 'traditional' reactor-safety code category. These are codes which were developed in the age of main-frames and card readers. Some of the input still harks back to those days by maintaining a fixed card type format. The output is designed to be printed and presents computational results in terms of labeled tables and parameter maps. The Fluent code, on the other hand, is a commercial package which has been developed for use on color X-windows, workstations and has to attract its customers in terms of 'user-friendliness'. Its input is entirely menu driven and grid generation is handled through a menu driven, graphical preprocessor. The output can be visualized using a color postprocessor, or it can be extracted and visualized using commercial plotting packages. Summarized in this fashion it would appear that the advantages are clearly with Fluent. On balance this is true, but Fluent does have its drawbacks. The Fluent menu driven input and grid generation can be confusing for a new user and the code manuals and tutorials are less help then could be expected. Furthermore, the menu type input forces the user to search through a multitude of unneeded options before locating desired features. The COMMIX code only requires the user to define those parameters required for a problem. However, the user must make sure that cards are placed in the proper order and either remember all the keywords or constantly refer to the manual.

In terms of output, however, there is no contest. Interpretation of the COMMIX output requires both patience and time, the issue is made even more cumbersome by the COMMIX requirement to specify the desired parameters and their location before running the problem.

Input/output issues are relevant, but they should not be overemphasized. Especially in a University environment the COMMIX code is an eminently usable (and economical) package, and it should be evaluated primarily on the merits of its mathematical characteristics. As Table I shows in this respect it offers a similar range of computational options as the commercial code for this type of problem. Here we note only those which have special relevance to the computation of boron dilution transients.

Computational Resource Issues

It has been noted that recent advances in this area have made CFD codes accessible in a University environment; however, this certainly does not mean that computational speed and memory requirements are not an issue. In our experience both aspects proved to be limiting and served to define both the detail of the noding representation and the number of problems which can be analyzed. A maximum noding structure of ≈40,000 nodes and a transient computation time of ≈5 sec proved to be close to the practical limit which can presently be attempted (the problems were run on time-share SUN Sparc-20 platforms). In this respect it is worth noting that the matrix solver options available in COMMIX proved to be very useful. For large problems the time advantages of the sequential over-relaxation (SOR) method was substantial (over a factor of 2).

For the Fluent code, the many options for reducing calculation times were not fully investigated. In general, the default code features were maintained. This includes the Power Law interpolation method and the SIMPLEC solver though underrelaxation factors were adjusted. In this preliminary investigation, the greater fidelity solutions provided by the higher order interpolation schemes were deemed unnecessary, and the associated increase in computational time was undesired. Convergence of the normalized residuals was left at the default value of 1×10^{-3} for the combination of pressure, velocities and mass fractions. The energy equation, although available, was not invoked in these simulations. A large degree of "trial and error" adjustments is needed in order to obtain a converged solution with Fluent. This is especially true when determining the underrelaxation factors to be employed and the time steps to be taken. By default Fluent solves the pressure and dilute component mass conservation equations using a multigrid (MG) block correction to the line gauss siedel (LGS) iteration technique. The LGS technique is applied to the remaining governing equations (velocity and turbulence parameters). Options for the solution 'sweep' direction, whether the code iterates from node 1 to node n or the reverse, were employed and multiple sweeps for each parameter were invoked. Convergence was seen to be greatly enhanced through the judicious choice of underrelaxation and block correction parameters. At this time the more advanced turbulence models available (Reynold's Stress Model and ReNormalization Group methods) in the Fluent code have not been tested. After final determination of the runtime parameters, the 2D simulations were carried out with 0.01 s time steps to 3.00 s, and the 3D with 0.01 s time steps out to 0.75 s. to s.. Output files were generated at 0.5 s intervals for the 2D cases and at .25 s and .75 s for the 3D cases. The files typically required 5 Mb storage space per file.

Geometric Characteristics and Model Representation

Figure 1 shows the geometry of the UMCP facility annular downcomer along with the corresponding TMI (prototype) geometry. For the purpose of this test series, the inner core barrel or the UMCP facility was modified to reproduce the major and also some of the secondary features of the flow path. The scaling approach required a match of the relative volumetric fractions of each major flow path segment (cold leg, downcomer, lower plenum) along with important flow geometry characteristics (e.g. ratios of the pipe diameter to hydraulic diameter of the downcomer).

As noted, the ratio of the flow channel increase in the test facility is equivalent to that of the prototype; thus, in a fully developed flow field the augmentation of turbulence and mixing produced by this feature should be comparable. However, the flow above the off set is not 'fully developed', the cold leg entrance occurs only ≈1.5 diameters above the location of the off-set, thus 'entrance effects' will certainly be present.

Using the Fluent code both 2D and 3D cases of the UMCP downcomer were developed and run. The noding structure for these cases is depicted in Figures 3 and 4. For both the "model" (UMCP) and "prototype" (B&W) geometries the inlet boundary condition to the cold leg was imposed as a diluted slug of liquid with velocity of 1 m/s. This velocity corresponds to nominal operation of one of the UMCP reactor coolant pumps (RCP) at rated shaft speed. The diluted fluid was given a density of 1300 kg/m^3 while the fluid initially present in the system was specified as 1000 kg/m^3. Identical viscosities (2.8×10^{-4} kg/m^2 s) were employed for both solutions along with a diffusion coefficient of 1×10^{-9} m^2/s [15]. Such high density differences are of course not probable under prototypical conditions. However, given the scaling and operating limitations (pressure/temperature) of the UMCP model a higher density difference must be employed to achieve Froude number similitude. Experimentally, such higher density differences are achieved with the use of concentrated salt solutions. The concentration difference is in addition to the fact that the diluted solution in the cold leg loop seals will be at a substantially lower temperature.

Boundary Conditions

The traditional 'wall' and 'flow continuity' type boundary conditions are available for both codes and perform as intended. For the representation of component segments of the type shown in Figure 2 additional boundary conditions are needed. Complicating features of the flow in the downcomer is that flow can occur both downward and radially. The preferred option is, of course, to model the complete circumference of the entire downcomer and the lower plenum. This modeling approach is used both by Alverez[12] and Gango[16]. For our initial study such an approach would have severely strained both available computational resources and time. Fortunately, since the primary goal of this study is the determination of relative rather than absolute flow field differences, the more limited geometric representation shown in Figure 2 was deemed adequate.

Sectioning of the downcomer raises the question of what boundary conditions are to be applied at the two faces of the radial segment. It is not proper to model an 'open' flow area because the flow resistance in the radial direction is higher. To model flow resistance both codes provide an option which adds an additional direction dependent resistance term to the momentum equation. The input details are different but the implementation is similar. Initially it appeared that in this respect the COMMIX code offers greater flexibility since it provides a direct input option for both volume and surface porosity. However, it turns out that this input does not influence the momentum equation but only the output of the velocity field. Thus, for both codes resistances had to be modeled by specifying imposed directional pressure gradients. This modeling feature provided both input and computational difficulties, and, at the time this report is written, the physical implications of boundary conditions are not satisfactorily resolved. In the 3D Fluent simulation of the UMCP facility only one cold leg and its associated 30° segment of the RPV are modeled. A radial resistance was imposed by placing a porous-media plane of cells adjacent to the outlet boundary conditions along the axial length of the downcomer. For the porous-media cells Fluent includes additional terms (sinks) to the momentum equations:

$$\frac{\mu}{\alpha}v + C_2\left(\frac{1}{2}\rho v|v|\right)$$

The factors α and C_2 are the permeability (m^2) and inertial resistance factors (m^{-1}) respectively in each component direction, which are specified by the user. For those models presented in this paper, the permeability was varied until a value was reached for which approximately 70% of the cold leg flow (by mass) exited at the bottom of the downcomer.

For two dimensional computations with the COMMIX code an approximation to the alternate flow path which exits in reality was simulated by providing a high resistance exit for the fluid at the top of the downcomer (Figure 2b). The imposed resistances were adjusted until the flow in the lower plenum represented ≈80% of the total flow.

Numerical Diffusion

Computations which seek to evaluate the degree of mixing during flow, have to confront the 'numerical diffusion' issue (Hassan[9], Jacobson[14]). For this particular study this requirement is less severe since the goal was not to determine the actual degree of expected mixing, but rather the difference generated by a specific change in flow path geometry. In this case it is adequate if: a) the numerical diffusion is not severe (to the extent that it overwhelms other phenomena), and b) it is similar for both analyzed cases. These requirements are fulfilled by both codes. As shown by the un-physical spread of the concentration wave front, for both codes numerical diffusion is present, but it is not sufficiently intensive and for the transient times in question, the wave front remains well defined. Note that, as shown in Table II, the COMMIX code offers the option of using a special finite difference scheme (the Skew Upwind Discretization Scheme) which is designed to reduce numerical diffusion for flow fields which are inclined to the mesh structure. The option was used in a test computation but did not show a noticeable reduction in the degree of numerical diffusion. This is to be expected, since in both of the modeled components, that is, in the cold-leg pipe and in the downcomer, the predominant flow is normal to the grid. The Fluent code does allow for higher order interpolation schemes (QUICK and Blended Second Order Upwind/Central Difference) in order to reduce numerical diffusion; however, these options were not tested. This was primarily due to the excessive time required to complete a simulation and workstation computing loads. The same reasons precluded a node size sensitivity analysis. Presence of numerical diffusion can be illustrated by the following example which used Fluent to analyze the mass fraction of salted solution along the cold leg centerline for the initial pump start transient at 0.1 seconds in a 2D representation of the UMCP model. At 1 m/s, the salted slug front could have only traveled 10 cm corresponding to the 26th node from the inlet boundary, yet concentrations an additional 10 cm downstream from this point are at a salted mass fraction of 0.014.

Computational Results

The computations had two principal goals:

1) To determine whether the opposing off-sets influence the downward flow resistance. This question is relevant since once flow from the cold leg reaches the downcomer plane, it has the possibility to move in several directions. The preferred path is downward into the lower plenum and then up through the core, however, flow can also move radially and subsequently enter one of the 3 other cold legs. Finally, if some of the RVVV's (Reactor Vessel Vent Valves) are fully or partially open, it has a limited possibility of moving upward. The question of what part of the clean slug of water eventually traverses the core, depends thus on the relative directional flow resistances which the flow encounters when it enters the downcomer.

2) To determine how the offset influences the distribution of the mixed slug front as it enters the lower plenum region.

One of the primary purposes of the 2D models was to provide experience in code operation and input/output options and to serve as a comparative benchmark between the two codes. For the COMMIX computation boundaries conditions at the upper and lower downcomer planes (Fig. 2) allowed a partial separation of the flow entering from the cold leg, on the other hand, flow in the 2D Fluent computation is restricted to downward flow. The 3D model was developed for a single UMCP cold leg and the associated 30° downcomer region. The geometry noding structure was defined in Cartesian indices and then body-fitted onto its proper cylindrical coordinates using the Fluent preprocessor (called Prebfc). The grid structures for the 3D simulations are shown in Figure 4. Although a total of 40,700 cells are used, the majority are wall or 'dead' cells leaving only 11,600 fluid cells.

In spite of the outlined differences, the various approaches yielded several consistent results. Thus, the answer regarding the first question, 2D results obtained from both codes show that the direction of the off-set does not alter total downward flow resistance. Thus, for the COMMIX code, the integrated fractional downward flow remains identical for both off-set directions, almost identical total downward flow rates are obtained also with the 2D and 3D Fluent computations. This implies that the integral flow of the boron-dilute volume into the RPV lower plenum (and by implication into the core) is not influenced by the direction of the off-sets.

On the other hand, the computed radial distribution of the slug front as it exits from the bottom of the downcomer, does reveal some off-set dependent differences. As shown in Fig. 5(a), for the prototypical geometry the radial distribution of the concentration front computed by the COMMIX code is peaked toward the core-barrel. In this calculation the entering slug is warmer by 10°C, and as seen, the COMMIX results show a clear radial temperature gradient. The 2D Fluent computations (Fig. 5(b)) show that the dilute solution enters the lower plenum at a slightly higher concentration nearer the RPV wall for the case of the offset being on the core barrel side of the downcomer This is borne out in the contour plots of Figure 6 for the 2D Fluent model at 0.5 seconds. There is no indication from these 2D results that would suggest that either geometry promotes greater mixing of the diluted water.

For the 3D Fluent simulation the question of how the off-set location influences the spatial distribution of the partially mixed slug front is considered in Figures 7 to 10. The contour plots presented in Figs 7 depict the salted mass fraction in the I-plane at a location 1 cell inward from the cold leg entrance for both cases (Figure 4). In general, the patterns which develop are similar for both offset types. For both the model and the prototype the penetration of salted fluid quickly develops into a double pronged front. Contours in the J plane through the centerline of the cold leg and downcomer are shown in Figure 8. These figures confirm the 2D results, which show a greater penetration of the dilute solution along the pressure vessel side of the downcomer in the case of the model geometry. The distribution of dilute fluid in the K-plane 10 nodes below the downcomer width change is shown in Figures 9 and 10. The initial 'double pronged' penetration of the dilute fluid into the downcomer is clearly evident in Figure 9 (as shown earlier in Figure 7).

The "double pronged" flow feature is caused by the imposed radial boundary conditions. The larger radial resistance is modeled by a localized porous medium which generates a step resistance retarding radial flow. In reality the higher radial resistances are more evenly distributed around the circumference of the downcomer. Thus the sharpness and location of this flow field feature is a computational artifact, however, the division of the downward plume into two roughly symmetric segments potentially could be present in reality. An indication of this is in the experimental work of Anderson[17] which shows that during a pump-startup transient the diluted plume enters the core in two separate fronts. In any case, the "channel off-set" feature analyzed in this study is cylindrically symmetric, thus they will not affect the azimuthal distribution of the plume.

Indeed, as shown in Figs 7 and 9 it appears with equal prominence for both off-set types. The important features are the radial distribution of the dilute slug as it penetrates the downcomer. As shown in Figures 8-10, there is a significant difference in the radial distribution. In the model the dilute slug preferentially penetrates the downcomer close to the core barrel, and in the prototype closer to the RPV. As mentioned earlier, this flow feature is also present in the 2D calculations. The differences in offset do produce a computable difference in the velocity and concentration profile of the dilute front.

Summary

An experimental program is underway at the University of Maryland College Park to investigate the generation, transport and mixing of boron-diluted water volumes in a B&W lowered-loop design PWR. In conjunction with this experimental effort, computational analyses are being conducted using the multi-dimensional COMMIX-1C and Fluent v4.2 CFD codes. One scenario of interest is that of a boron-dilute volume of water being transported through the cold leg, downcomer, lower plenum and eventually into the core regions as a result of a reactor coolant pump start. The experimental facility scaling necessitated modifications to the downcomer geometry which are not typical of the prototypical layout. To asses the effect of these geometrical differences upon boron mixing, two and three dimensional CFD simulations of the UMCP facility geometry have been developed and run. The results show that for the UMCP, or model, geometry the effect of having a widened downcomer gap on the core barrel side does not result in significantly greater mixing of the dilute solution although it does affect the location of the plume in the downcomer width. Two-dimensional calculations using the COMMIX-1C code were completed and showed no noticeable differences in mixing for the two types of geometries simulated. Key aspects of this study were the 'ease of use' and computational models associated with these two codes given the university environment in which they are employed. Although the user features of the Fluent code make it more amenable to model generation and result visualization, its multitude of options, menu structure and documentation can lead to confusion, and did, for a new user.

This work is being performed under USNRC contract NRC-04-94052

References

[1] Green, J., K. Almenas, and M. di Marzo, *Generation, Transport and Mixing of Boron-Dilute Water in a Scaled Integral Model of a B&W Reactor*, Submitted to CSNI Specialist Meeting Sub of WRSIM on Boron Dilution Reactivity Transients, State College, PA, October, 1995.

[2] Anderson, N.G., Hemstrom, B., Karlsson, R., and Jacobson, S. Physical Modelling of a Rapid Boron Dillution Transient, 1995.

[3] Diamond, D.J., Kohut, P., Nourbakhsh, H., Valtonen, K., and Secker, P. *Probability and Consequences of a Rapid Boron Dilution in a PWR*, NUREG/CR-5819 (BNL-NUREG-52313), 1992.

[4] Nourbakhsh, H.P. and Cheng, Z., *Potential for Boron Dilution During Small Break LOCAs*, NUREG/CR (BNL Technical Report W-6156), October, 1994.

[5] Sun, J. G. and W. T. Sha, *Analysis of Thermal Mixing and Boron Dilution in a PWR*, NUREG/CR-5822 (ANL-91/43), 1993.

[6] Tinoco, H., B. Hemström, and N. Andersson, *Physical Modelling of a Rapid Boron Dilution Transient*, VS-S 93:B21, Vattenfall AB, Sweden, September, 1993.

[7] *Fluent Version 4.2 Users Guide*, Fluent Incorporated, Lebanon, NH, 1993.

[8] Domanus, H. M., et. al., *COMMIX-1C: A Three Dimensional Transient Single-Phase Computer Program for Thermal Hydraulic Analysis of Single and Multicomponent Systems*, NUREG/CR-5649, Argonne National Laboratory Report ANL-90-33, September, 1990.

[9] Hassan Y. A. "A Three-Dimensional Thermal and Fluid Mixing Analysis Using the Mass-Flow-Weighted Skew-Upwind Differencing Scheme", NP-3870, February, 1985.

[10] Theofanous, T.G. and Lyer, K., *Mixing Phenomena of Interest to SBLOCAs*, Nuclear Engineering and Design, Vol. 102, pp 91-103, 1987.

[11] Theofanous, T.G., LaChance, J.L., and Williams, K.A., *The Thermal Hydraulics of SBLOCAs Relative to Pressurized Thermal Shock*, Nuclear Science and Engineering, Vol 102, pp 74-100, 1989.

[12] Alverez, D., Daubert, O., Janvier, L., and Schnieder, J.P., *Three Dimensional Calcualtions and Experimental Investigations of the Primary Coolant Flow in a 900 MW PWR Vessel*, Proceedings of NURETH-5, Salt Lake City, Utah, 1993.

[13] Alverez, D., Martin, A., and Schnieder, J. P., *Boron Mixing Transients in a 900 MW PWR Vessel for a Reactor Start-up Operation*, Proceedings of NURETH-5, Salt Lake City, Utah, 1993.

[14] Jacobson, S., *Risk Evaluation of Local Dilution Transients in a Pressurized Water Reactor*, Ph.D. Dissertation - Department of Mechanical Engineering, Linköping University, No. 275, 1992.

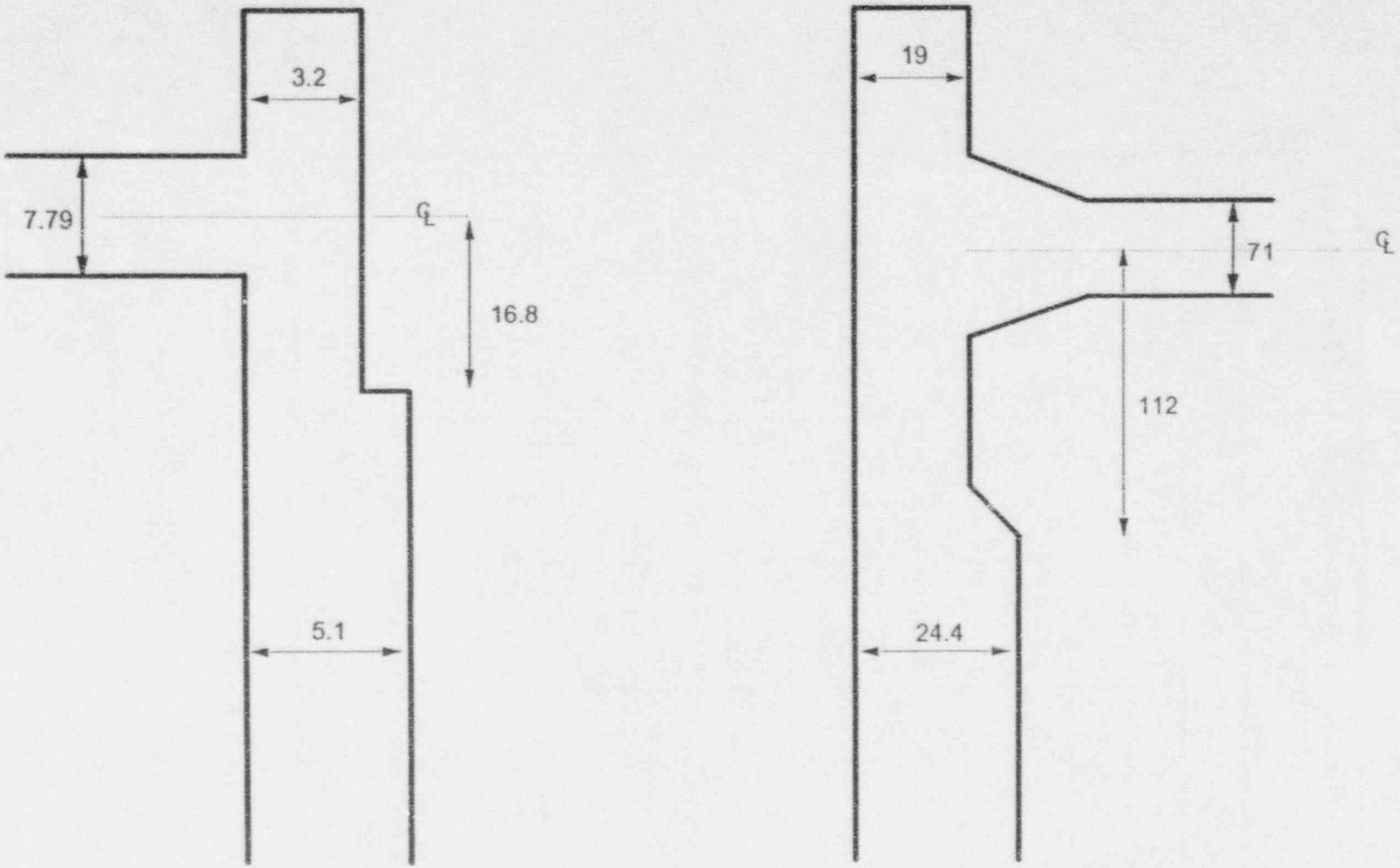
[15] *Chemical Engineers' Handbook*, McGraw-Hill, New York, 1973.

[16] Gango, P., *Numerical Boron Mixing Studies for Loviisa Nuclear Power Plant*, Procedings of NURETH-7, Saratoga Springs, New York, 1995.

[17] Andersson, N.G., Hemström, B., Karlsson, R., and Jacobson, S., *Physical Modelling of a Rapid Boron Dilution Transient*, Procedings of NURETH-7, Saratoga Springs, New York, 1995.

Table I - Code Comparisons : Calculational Models		
Feature	Fluent v4.2	COMMIX-1C
Dimensionality	2D / 3D Cartesian, Cylindrical, Curvilinear. Allows Body Fitting	2D / 3D Cartesian
Fluids / Phases	Incompressible & Compressible, Dispersed 2nd Phase	1 Phase Incompressible
Solution Methods	Control Volume Finite Difference, Non-Staggered	Control Volume Finite Difference Skew-Upwind Staggered
Solver	SIMPLE, SIMPLEC, or GMRES	SIMPLEST-ANL
Interpolation	Power Law, QUICK, or Blended 2nd Order Upwind Difference	--
Turbulence	k-ε, RSM, RNG	k-ε

Table II - Code Comparisons : Computational Requirements		
Feature	Fluent v4.2	COMMIX-1C
Workstation	SUN Sparc-20	SUN Sparc-20
Available RAM	114 MB	114 MB
Operating System	SUN-OS	SUN-OS
Input	Grid generated using pre-processor. Menu driven input tables. Some difficulty in specifying options	ASCII input file
Output	Allows graphical display of geometry and calculated variables. Allows integration of various flow planes. Numerous display options.	Requires pre-determination of variables to be output. ASCII output files.
Sample Problem: 2D Model Geometry	25,000+ nodes 5 s Transient ~18 hrs	10,000+ nodes 3 s Transient ~ 2 hrs
Sample Problem: 3D Model Geometry	40,000+ nodes 0.75 s Transient ~ 20 hrs	not done



(a) MODEL - UMCP

(b) PROTOTYPE - TMI

Figure 1 - Mixing in Annular Downcomer Region

(not to scale - dimensions in cm)

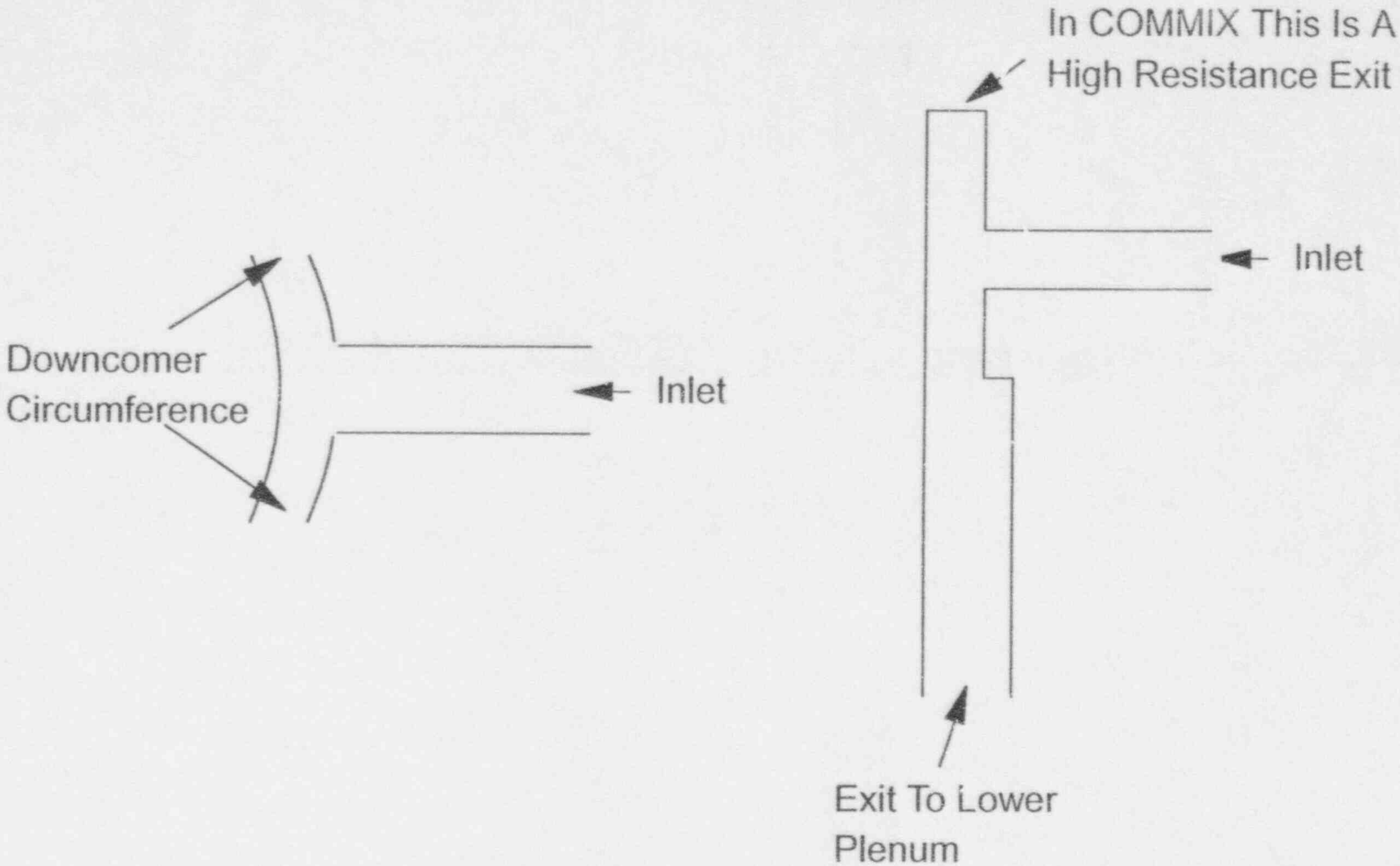


FIGURE 2: 2D and 3D Flow Boundaries

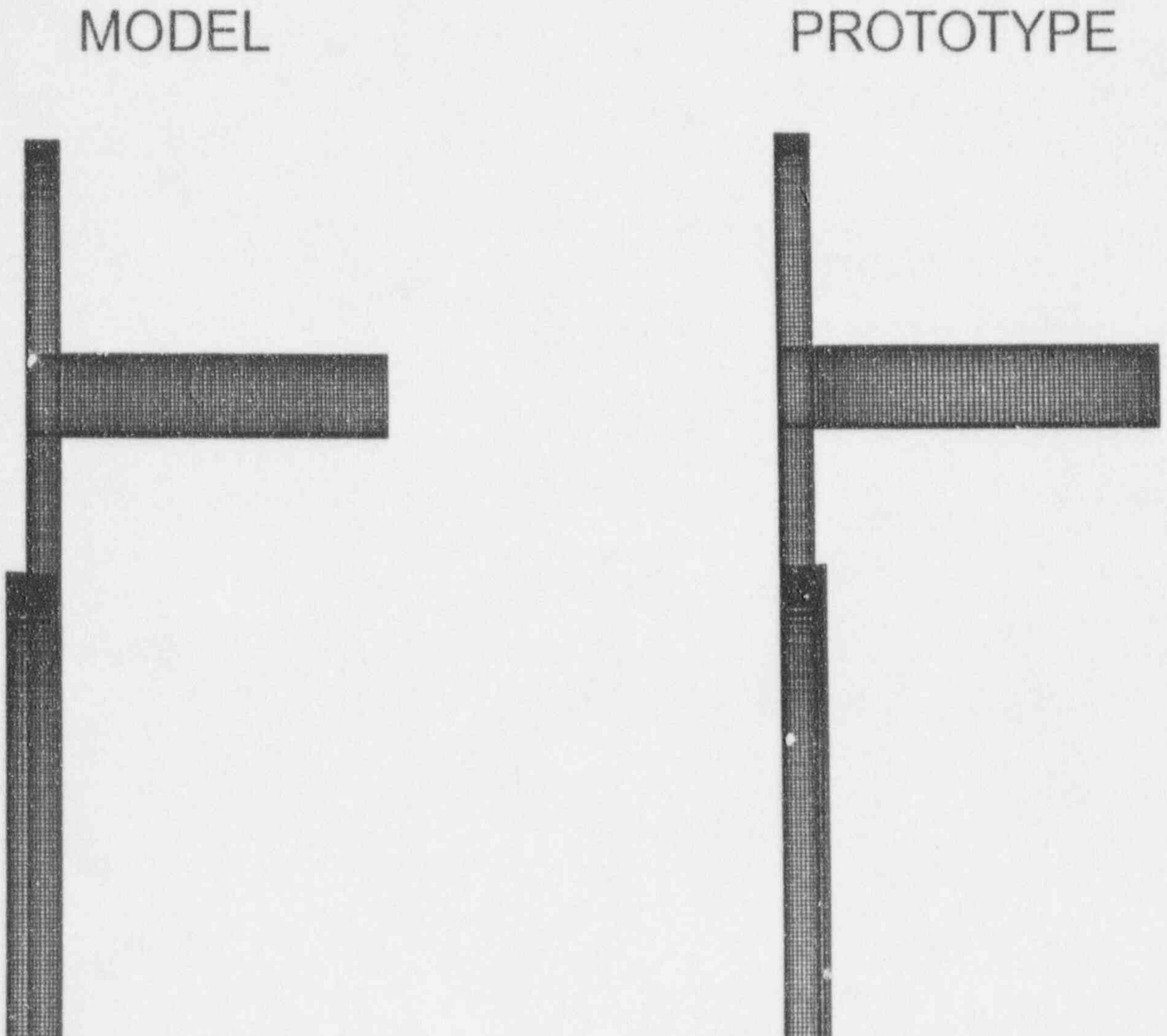


FIGURE 3: 2D GRID

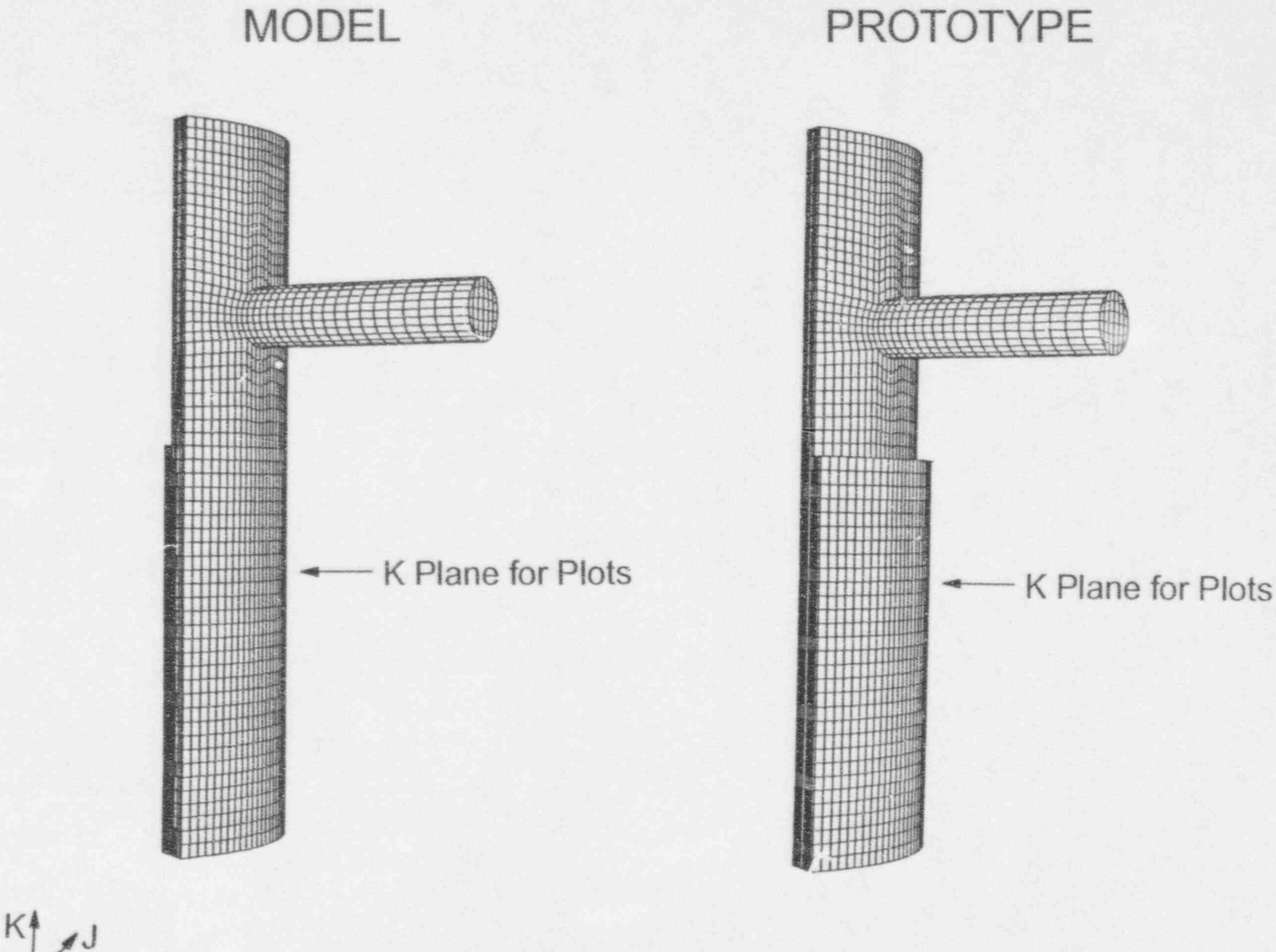
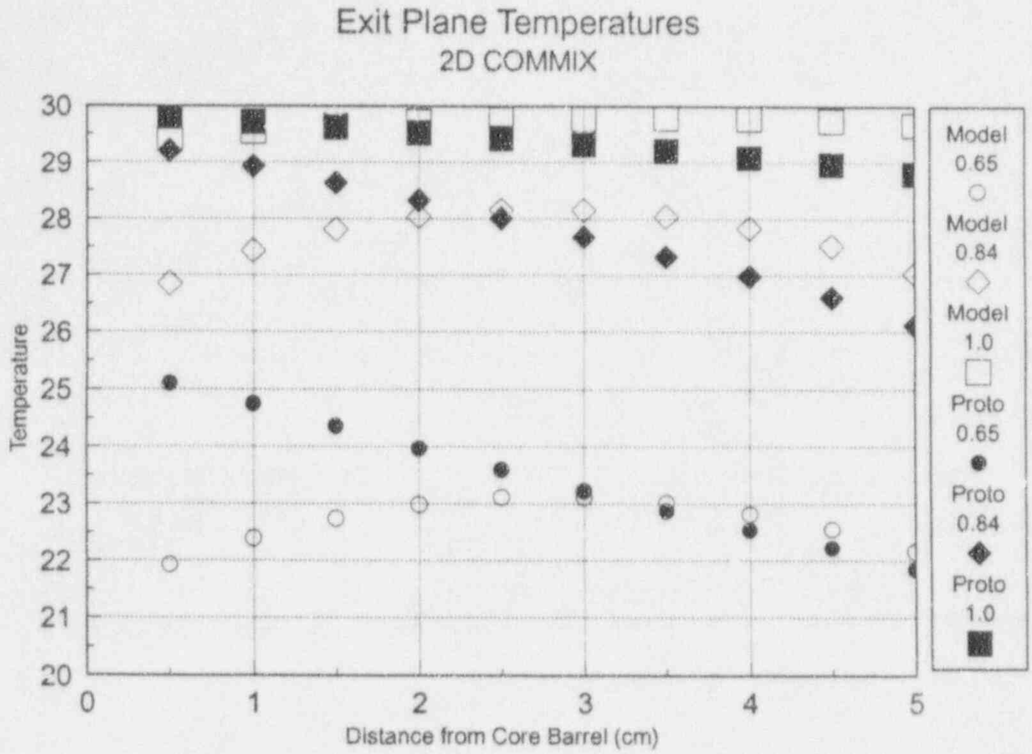


FIGURE 4: 3D GRID



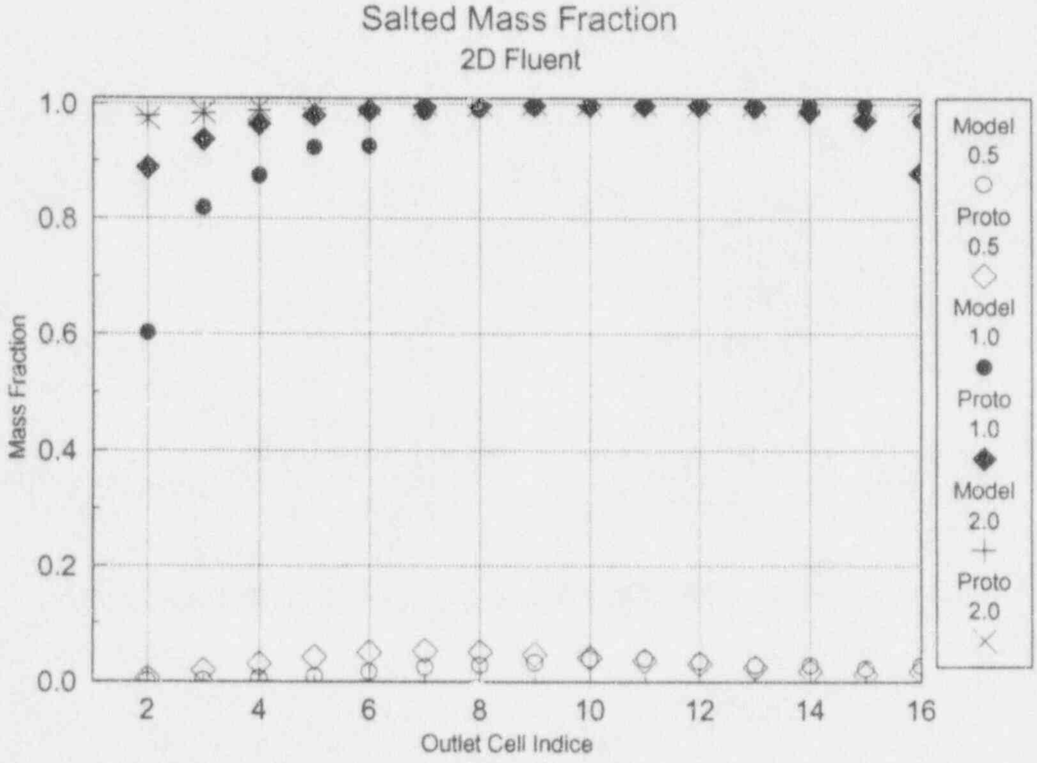


FIGURE 5: 2D CODE RESULTS

2D SALTED MASS FRACTION

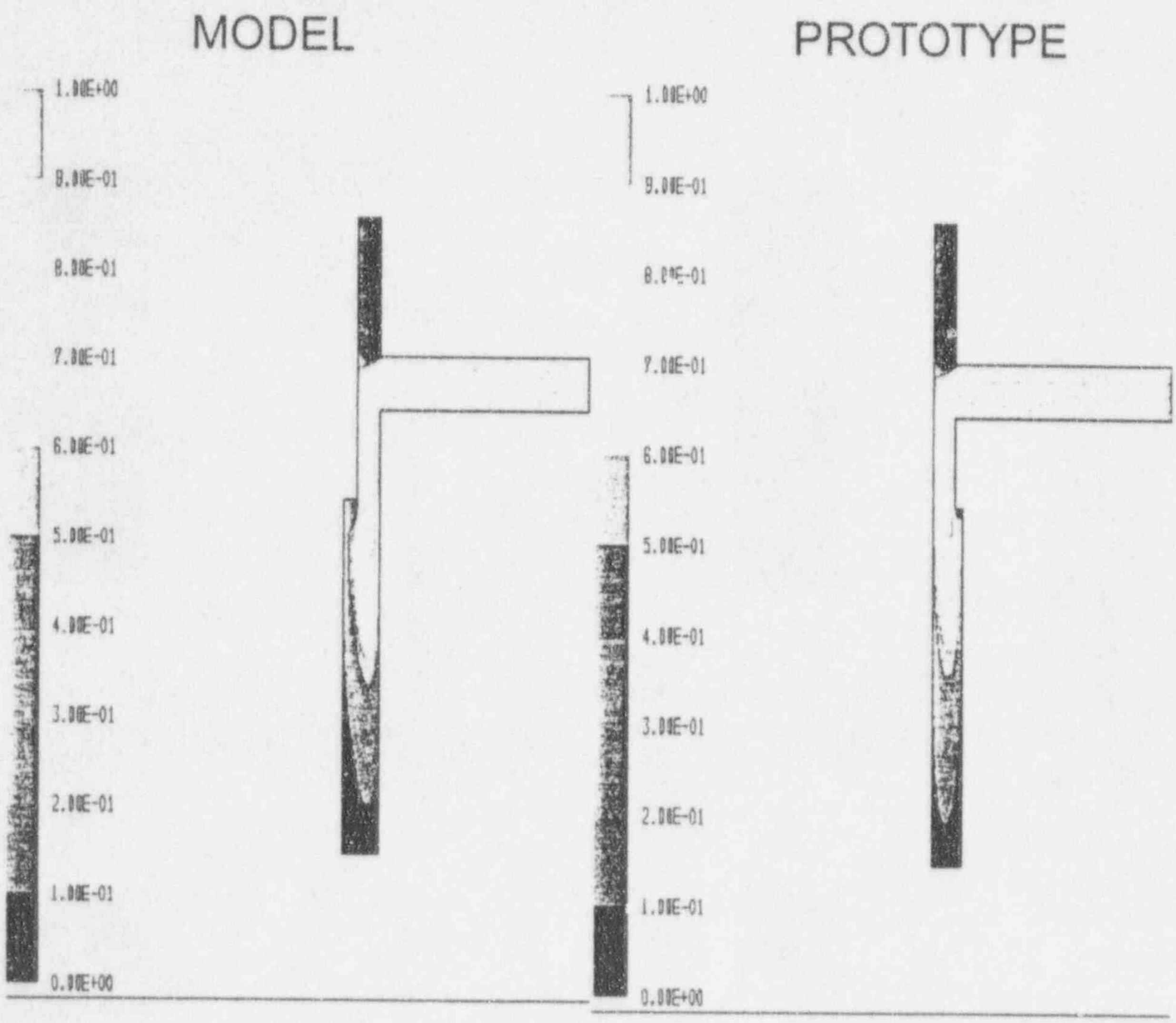


FIGURE 6

3D SALTED MASS FRACTION
ONE I-PLANE FROM INLET

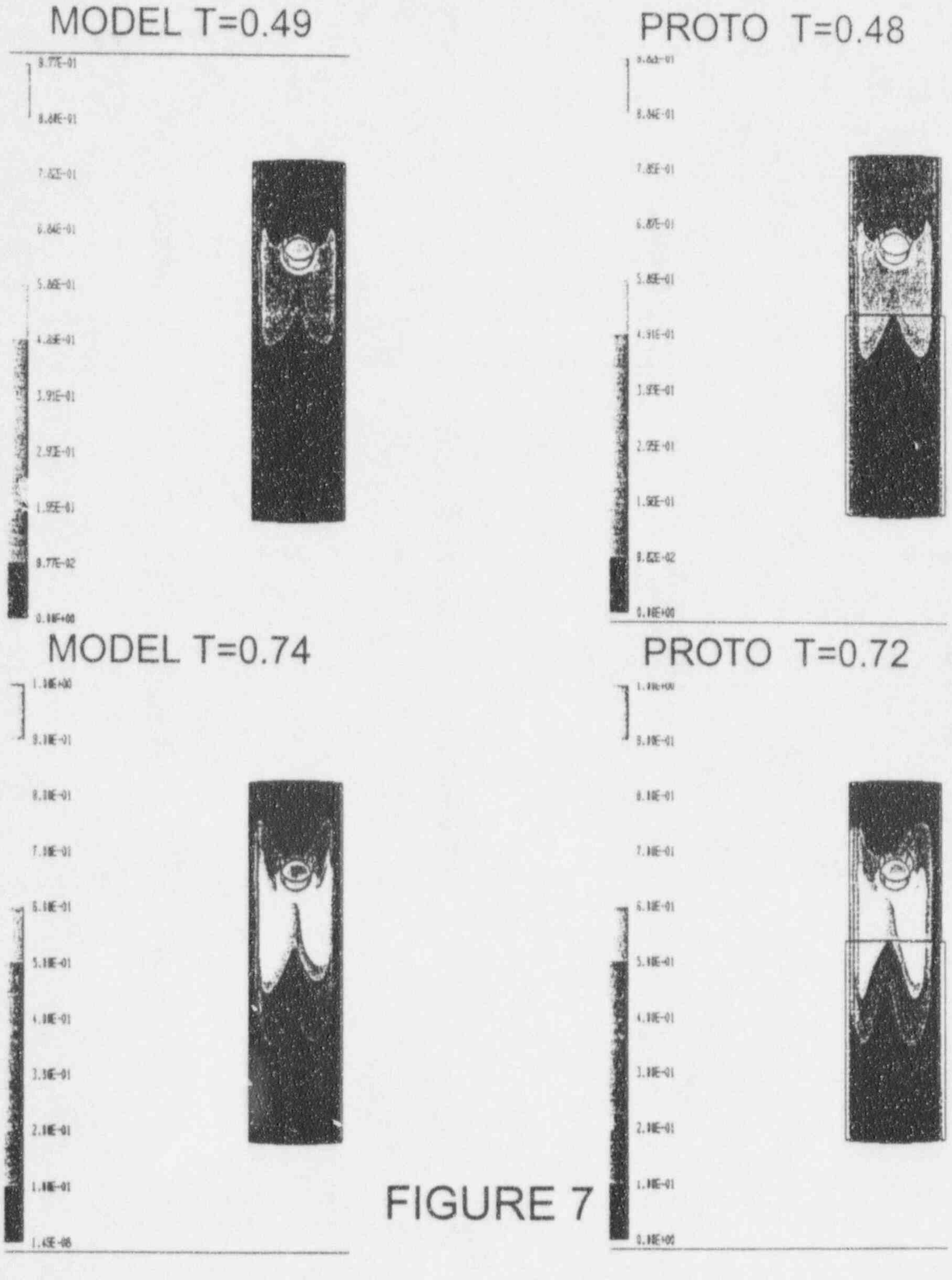


FIGURE 7

3D SALTED MASS FRACTION
CENTERLINE J-PLANE

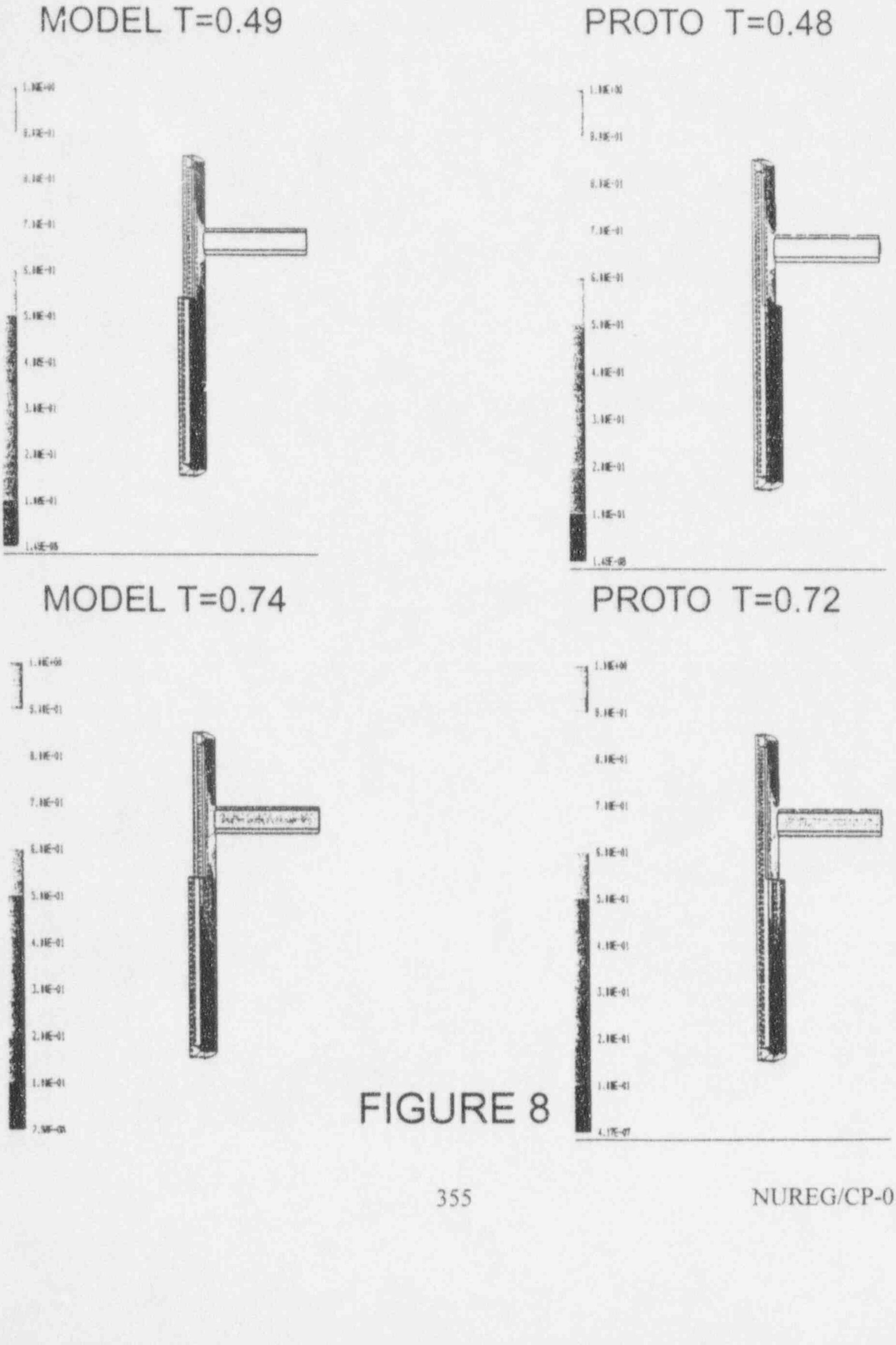


FIGURE 8

SALTED MASS FRACTION
10 K NODES BELOW WIDTH CHANGE

MODEL T=0.49 PROTO T=0.48

MODEL T=0.74 PROTO T=0.72

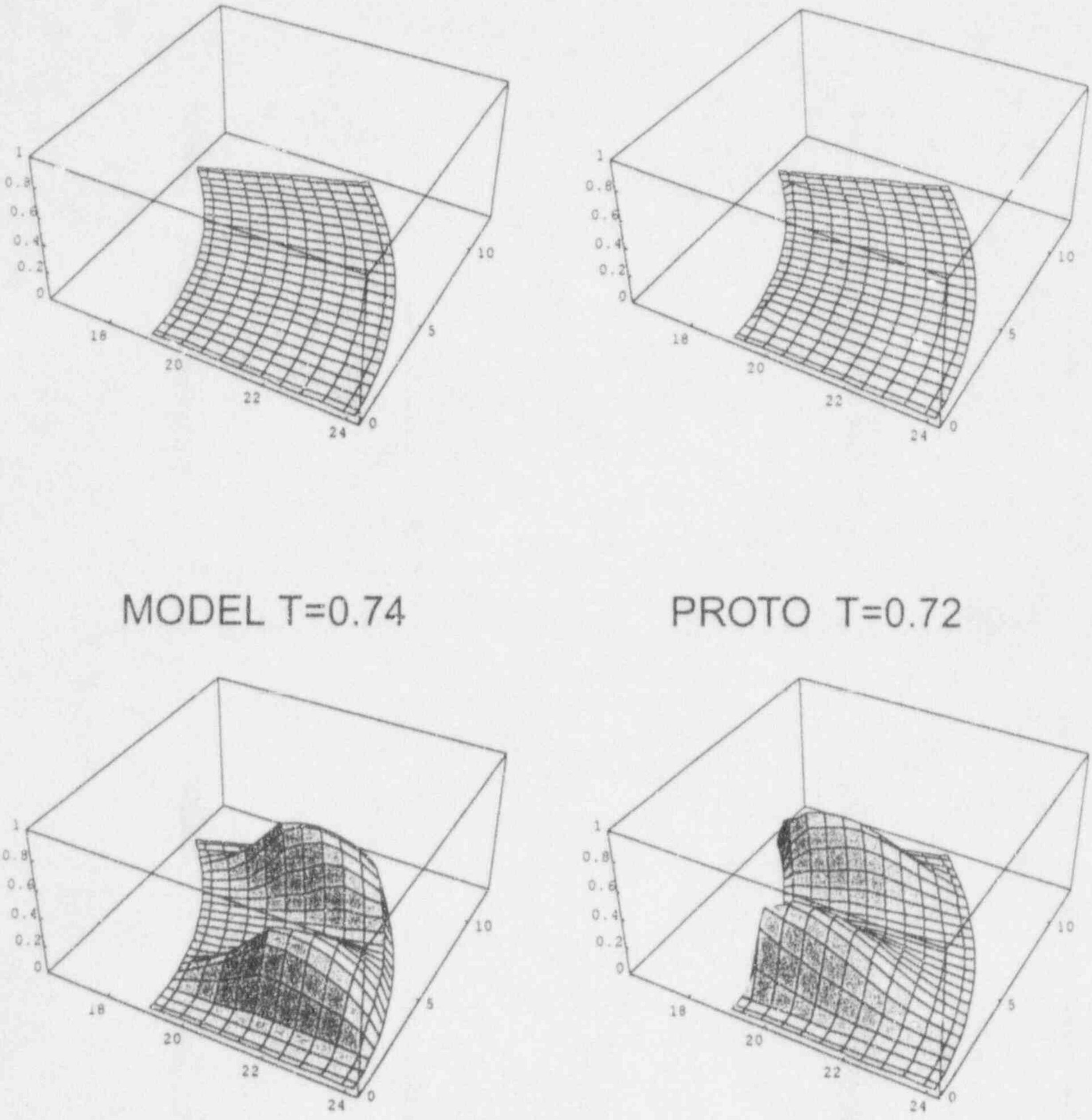
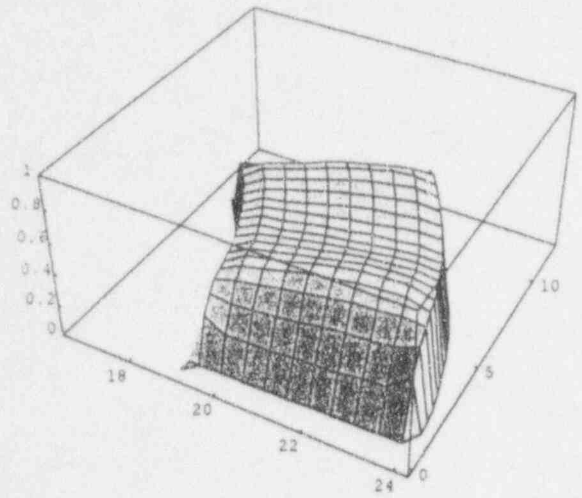


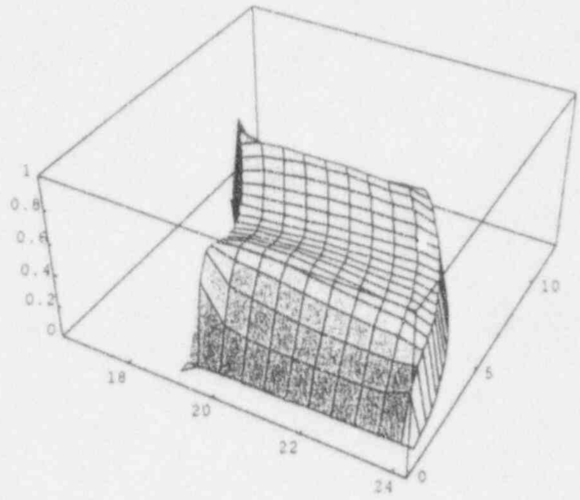
FIGURE 9

-K VELOCITIES
10 K NODES BELOW WIDTH CHANGE

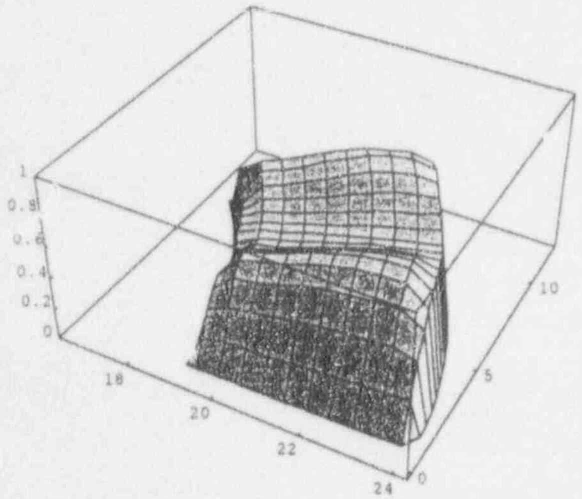
MODEL T=0.49



PROTO T=0.48



MODEL T=0.74



PROTO T=0.72

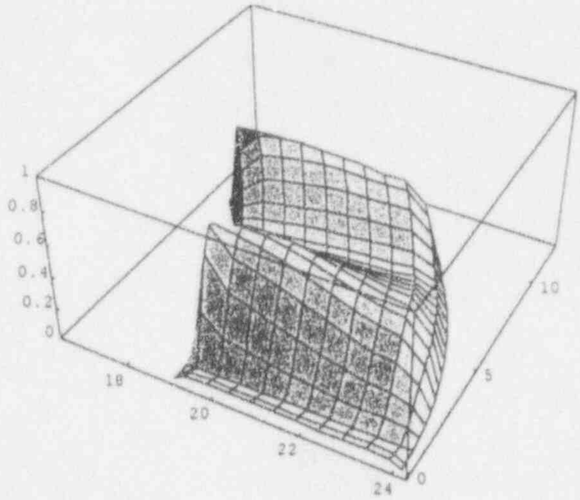


FIGURE 10

Generation, Transport and Mixing of Boron-Dilute Water in a Scaled Integral Model of a B&W Reactor.

J. Green, K. Almenas, and M. di Marzo[†]
Nuclear & Materials Engineering Department
[†]Mechanical Engineering Department
University of Maryland
College Park, MD 20742

ABSTRACT

The University of Maryland College Park (UMCP) 2x4 thermal-hydraulic test facility is being used to assess the generation, transport and mixing of boron-dilute water volumes. The facility is a reduced height / reduced pressure scaled model of the B&W lowered loop plant and includes all relevant features of the prototype. This paper describes the UMCP facility scaling rationale, the instrumentation layout and program test matrix. Initial results from tests conducted at the UMCP facility are presented.

INTRODUCTION

From the time that the use of soluble neutronic poisons in the primary system were initially proposed, great care has been exercised to reduce the possibilities of an inadvertent poison dilution. One of the principal physical features which greatly reduce the probability of such an event is the large volume of water in the primary system (on the order of 300 m^3 for a typical PWR). The rate at which boron-free coolant could be added is ultimately controlled by piping sizes and feeding devices. As a consequence, whenever the primary system is active, the added boron-free water mixes with the borated inventory and the resulting boron-dilution transient is sufficiently slow that it can be readily diagnosed and controlled.

Recent developments have spawned new interest in boron induced reactivity transients. First, the trend towards higher burnups and longer times between outages has gradually forced up the soluble boron concentration used in the plants. In some cases it now approaches and exceeds 3000 ppm, while during the design phase boron concentrations in the 2000 ppm range were considered. Second, though it had very different causes, the reactivity initiated Chernobyl accident led to a thorough re-examination of all possible reactivity events. Recently a number of investigators[1,2,3] have pointed out that for certain conditions, it is hypothetically possible for relatively boron free water volumes to accumulate in components of the primary system. Necessary preconditions are: a) the primary system is completely (or almost completely) stagnant and b) a mechanism is present which either separates or adds boron free water to the primary system. Specific boron dilution transients have been analyzed using single dimension[4] and three dimensional[5,6], finite difference codes. Many of these computational investigations have concluded that experimental studies are needed, especially in order to reliably quantify the buoyant and/or turbulent mixing of boron diluted water slugs as they move in the cold legs and the downcomer region.

A potential separation mechanism is available if the system is operating in the Boiler-Condenser Mode (BCM) for OTSG plant designs or in reflux-condensation mode for U-tube design steam generators. This can occur under SB-LOCA conditions when sufficient primary inventory has been removed such that the upper regions of the system are steam filled. Since boric-acid is relatively insoluble in steam, the boiling-condensing operation will generate an almost boron free condensate which can accumulate in the cold leg loop seals or, for the case of once-through steam generators plants, in the steam generators. Subsequently, when motion of water is re-initiated, the boron-free water volume can be transported to the core. The extent to which this diluted water mixes while it is being transported to the core region, is one of the issues which will be investigated in this series of tests.

It is worth noting that potential precursor events have been observed in operating plants. Of note is the June 6, 1993 event at the San Onofre PWR[7] which showed that with pumps inactive and a low level of decay power, it can take on the order of days before boron solutions are uniformly distributed throughout the system.

The UMCP facility is a reduced height / reduced pressure scaled model of the B&W lowered loop plant. Volume and power are scaled by a factor of 1/500 while linear heights are reduced by 1/4.4 The model includes all relevant features of the prototype, including reactor pressure vessel with annular downcomer, 4 cold legs with loop seals and reactor coolant pumps, 2 'candy-cane' type hot legs and once through steam generators (OTSG), as well as a pressurizer and associated auxiliary systems. A schematic of the UMCP facility is shown in Figure 1. The facility has been employed in the past to investigate SB-LOCA phenomena[8,9,10], loss of residual heat removal, and more recently, natural circulation above a degraded core[11].

The UMCP test program can be divided into two inter-related parts:

1) The accumulation or generation phase, which proceeds under stagnant flow conditions. This determines also the range of diluted coolant volumes, the relative concentrations of diluted and undiluted coolant and the state of the primary system (i.e. pressure, liquid levels and distribution of temperatures).

2) The 'diluted volume transport phase'. This part depends on the motion initiating mechanism. Mechanisms considered include, Reactor Coolant

Pump (RCP) start-up, loop clearing, initiation of natural circulation through ECCS addition and mechanisms unique to OTSG plants, such as the Interruption Resumption Mode (IRM). Natural circulation can be reestablished either by ECCS introduction or changes occurring on the secondary side which lead to the initiation of an IRM operation. The later mode is one of the characteristic modes of the OTSG type plants.

The accumulation phase is driven either by external effects (e.g. leakage of clean water through pump seals), or is a characteristic of the entire system (e.g. accumulation of distillate during BCM operation). The UMCP facility previously operated in the BCM and no modifications are required to reproduce this mode. Scaling issues are reduced to the specification of heater-power operation-time combinations which can represent the expected envelope of accumulated condensate in the prototype.

It is the second test phase which leads to modifications of some components in the 2x4 Loop facility and therefore require guidance from scaling relationships.

For inertially driven initiating mechanisms, the influence of density differences on fluid mixing has only second order importance. In fact, an experimental program under way in Sweden[12,13] chooses to ignore them altogether. The mixing is dominated by turbulent eddy diffusion and by entrainment occurring in those regions of the flow path where flow direction changes occur (e.g. the entrance to the downcomer, and the lower plenum of the pressure vessel). Modifications to the UMCP Facility include those which simulate the widening of the downcomer present in the prototype, the obstructions presented by the core barrel lugs and the instrument tube obstructions which penetrate the lower plenum. Special care is taken to assure that the merging of the downcomer and lower plenum components present a similar inertial resistance.

For slowly initiating flows an important scaling issue is the nature of the buoyancy force. Prototypical fluid density differences can have two origins: boric acid concentration and temperature differences. The effect that the addition of H_3BO_3 has on the density of water, is shown in Figure 2 superimposed upon the density difference caused by a range of temperature differences. Even at the high boron concentration of ~3500 ppm the figure shows that density differences will be controlled by temperature differences. If the density differences are caused by the H_3BO_3 concentrations alone (that is, the streams have equal temperature) then the Fr numbers would be so large that mixing is assured. The mixing tendency is dampened when density differences are large and Fr numbers are reduced. Thus the conditions which must be investigated with special care are those for which the borated and un-borated liquid volumes have large temperature differences.

For prototypical conditions temperature differences between the mixing volumes can be large. The maximum liquid temperature differences which could hypothetically occur in the prototype cannot be matched in a reduced pressure facility. However, since mixing depends on the relative density difference, this difference can be enhanced by using a soluble salt ($NaNO_3$) in the scaled facility. This approach will be used in the UMCP facility in order to bring the Fr number values into prototypical range.

While some of the investigated issues are plant-type specific (e.g. the possibility of accumulating boron free water in the steam generators and flow initiation by the IRM), others have a wider generality. Table 1 denotes the three fundamental aspects of the research program and gives a list of detailed items affecting each.

Table 1 - UMCP Boron Dilution Experiments

Generation	Transport - Initiation	Mixing
· Boiling-Condensing Mode after SB-LOCA · Improper ECCS / CVCS Injection · RCP Seal Water · Secondary to Primary Leak	· RCP Restart · Re-establishment of Natural Circulation · Interruption-Resumption Mode Operation · Large Vapor Space Condensation Event · ECCS / CVCS Injection	· SG Outlet & Cold Leg Loop Seals · Jet Impingement in Downcomer Annulus · Change in Downcomer Gap Width · Lower Plenum Guide Lugs and Instrumentation · Flow Distributor

SCALING & DESIGN OF UMCP FACILITY

The problem at hand is to provide a reliable quantification of the degree of mixing which occurs when two fluids, having either very similar, or somewhat different densities, flow along a specific, relatively complicated flow path. The nature of the motive forces and the characteristics of the flow path place the boron mixing problem in the general category of confined jet and/or plume induced mixing.[14,15]

A very similar type of problem in the same flow geometry was encountered during the earlier resolution of the pressurized thermal shock (PTS) issues.[16-21] In this case, the question posed was: to what extent does cooler water, which is introduced during actuation of ECCS, mix with the hotter loop flow? A scaled testing program was conducted to provide verification of issue resolution using test facilities having scale ratios of 1/5 (Creare[22]), 2/5 (Finland[23]), 1/3 (Japan[19]) to ½ (Purdue[17]).

Though the relationships derived from these programs differ in detail, they all agree on the premise that for the flow velocities and density differences encountered in nuclear systems, mixing phenomena of fluids having different densities are best correlated by equating ratios of inertial and buoyancy forces. The most widely used correlative index of this ratio is the Froude (Fr) number, and its squared inverse, the Richardson (Ri) number:

$$Fr = \frac{V}{[gL\frac{\Delta\rho}{\rho}]^{0.5}} \qquad Ri = \frac{gL\Delta\rho}{\rho V^2} \tag{1}$$

Where V is an average velocity, L a characteristic length and the density difference ratio refers to the densities of the mixing fluids.

A recognized potential weakness of tests performed using Fr number scaling is that for reduced scale facilities the prototypical Reynolds number (Re = V L/ν) cannot be reproduced simultaneously with the Fr number. This has to be considered especially for cases where the prototype and test facility flow conditions bracket the laminar-turbulent transition. For most conditions, at which test facilities are operated, this transition can be avoided.[13,19] As long as the Re number is above ~$1x10^4$, its variation, even over a considerable range, has a small effect on measured mixing rates in simulated downcomer geometries.

Boric acid dilution events are restricted to a narrower range of operating conditions than those for which PTS is possible. The buildup of appreciable volumes of boric acid free coolant requires a stagnant, or nearly stagnant primary system, whereas PTS stresses could be imposed even when significant loop flow is present. This circumstance has to be considered when applying the 'Regional Mixing Model' concepts[17,24] developed in the PTS study program to boric acid dilution events[25].

For many reactor related thermal-hydraulic problems the most powerful scaling tools are the extensively verified system codes like RELAP5 or TRAC. However, because of the nature of their mathematical representation, integral system codes can provide only very approximate indications of mixing. At best, they can be used to establish boundary conditions for more detailed calculations[4,26]. It is apparent that mixing related problems will have to use analytical tools which are able to represent at least 2 dimensional, and preferably 3 dimensional fluid flow fields. Of note is the development of finite difference schemes which reduce artificial mixing tendencies produced by numerical diffusion[20]. Several efforts have been undertaken to evaluate mixing phenomena in primary system components using 3D CFD codes[3, 5-6, 36-38]. A recently completed analysis has shown that CFD codes can be adapted to encompass the entire primary system[5]. It is clear that in the final resolution of the boric dilution problem, CFD will have a significantly larger role than it did for the PTS issue.

The use of simplified indexes as the Fr number in the scaling of complicated flow geometries requires several additional considerations. One of the cautionary issues is that, the definition of a characteristic length is not straightforward. In the problem of interest, the relevant flow geometry encompasses the lower plenum of the steam generator, the cold leg pipe, the annular downcomer and the lower plenum of the reactor pressure vessel. In order to ensure that a test facility-to-prototype Fr number ratio evaluated, say in the cold leg (where L is the inside diameter), applies also in the downcomer and the other flow components, it is necessary to **maintain linear scaling of component dimensional characteristics**.

The second requirement is to ensure that, as fluid conditions change during the transient, the prototype and test facility density ratios changes follow the same relative trend. **This requires that the component volumes of the test facility are in the same ratio as those in the prototype**.

The above two principles provide the main scaling guidelines. In addition, care is taken to ensure that as far as possible a) secondary flow path irregularities, (e.g. restrictions or widening of the flow path) which are present in the prototype, are reproduced in the test facility and, b) the resistances of alternate system flow paths maintain a comparable ratio with the primary flow path.

Information regarding the original geometry of the UMCP Facility can be found elsewhere[27-30]. Based upon the originally constructed facility geometry it was evident that to maintain scaling of component volume ratios then the model downcomer and lower plenum volumes must be increased while the core volume is decreased. Given the existing core and pressure vessel design, this was accomplished by decreasing both the length and radius of the UMCP core barrel. The remaining components, are less important to the mixing study at hand and are adequately scaled. While modifying these component volumes, the linear dimensions of the flow path and the features which generate turbulence where designed to be analogous with prototypical values.

A significant flow path difference between the prototype and the unmodified test facility can be observed by noting the junction between the downcomer and the lower plenum in Figure 3. As seen, the bottom of the test facility vessel is elliptical, while the TMI-2 vessel is hemispherical. This relevant flow path feature was simulated in the test facility by an appropriately shaped insert (Figure 5) which effectively changes the dish-shaped UMCP pressure vessel into a more hemispherical geometry. This is an important feature since the flow path of a diluted volume of water as it traverses the downcomer and lower plenum will be strongly affected by this facet of the prototype design.

The implemented modifications both shortened the core and reduced the diameter of the core barrel. The modified component volume fraction ratios are presented in Table 2. The later change made it possible to simulate an additional flow path feature which is shown more clearly in the schematic of Figure 6. That is, in the prototype the vessel wall bellow the cold leg entrance is thinner. This produces a non-minor widening of the downcomer gap. As shown in the figure this widening is simulated also in the test facility, however, since it was not possible to modify the test facility RPV wall, the recess occurs on the core barrel side. An independent program is underway at UMCP to utilize three-dimensional computational fluid dynamics (CFD) codes such as Fluent and COMMIX to assess the influences which differences between prototype and model geometry have upon mixing behavior[31].

Table 2 - Volume Fractions of UMCP & Prototype - After Modifications

	Model - UMCP		Prototype - TMI-2		
Region	Volume (m^3)	Fraction of Total	Volume (m^3)	Fraction of Total	Ratio P/M
Cold Legs	0.077	0.129	41	0.136	1.08
Downcomer	0.055	0.092	28.3	0.094	1.02
Lower Plenum	0.033	0.056	17.6	0.058	1.04
Core	0.055	0.093	22.1	0.073	0.79
Upper Plenum	0.05	0.084	30.9	0.102	1.21
Upper Head	0.035	0.059	15.6	0.052	0.88
Hot Legs	0.057	0.096	27	0.089	0.93
Steam Generators	0.233	0.392	119.6	0.396	1.01
Total	0.595	1	301.9	1	--

In addition to achieving appropriate component volume fraction similitude between the prototype and scaled model, it is desirable to achieve a correspondence between linear dimensions such as those presented in Table 3.

Table 3 - Characteristic Dimensional Ratios			
Item	Prototype TMI	Model UMCP	Ratio - P/M
Upper Downcomer Gap to Upper Downcomer D_h	0.45	0.50	0.90
Lower Downcomer Gap to Lower Downcomer D_h	0.50	0.50	1.00
Cold Leg Diameter to Core Barrel Diameter	0.19	0.19	1.00
Upper Downcomer Gap to Lower Downcomer Gap	1.3	1.5	0.87
Number of Cold Leg Diameters to Gap Change	1.6	1.9	0.84

The table shows that the major dimensions in the region where cold leg flows enter the annular downcomer are well reproduced by the UMCP Facility. For example, consider fully developed flow in the cold leg impinging upon the curved reactor core barrel it is seen that the ratio of jet diameter to core diameter is maintained. In addition, the ratio of gap thicknesses to hydrau[illegible] downcomer flow (D_h), both above and below the transition point, are linea[illegible]portional. The ratio of change in gap thickness between the prototype and model is nearly reproduced (0.87) although, as stated, this change of width is made in opposite directions. A final dimension of importance is the number of hydraulic diameters from the point at which flow enters from the cold leg until the transition point where downcomer gap thickness increases. The model therefore allows a greater length for flow from the cold leg to develop prior to reaching the transition point in this respect.

One of the scaling goals is to reproduce the flow irregularities present in the prototype as adequately as feasible. One major flow mixing feature of the B&W plant are the in-core instrumentation nozzles which enter the pressure vessel from the bottom and extend into the core through the flow distributor in the lower plenum. The TMI-2 design incorporates 47 incore instrumentation guide tubes in the lower plenum. These tubes are an obstruction to flow, and therefore their impact is simulated by reprocessing the same ratio of obstructed flow area to total core flow area (see Figure 4).

Other mixing features from the prototype are the lower plenum guide lugs and the upper core barrel 'horseshoe baffle'. Both of these features have been provided for in the UMCP Facility design.

It is also desirable to reproduce similar resistances to flow in the alternate flow transport circuits. For example, after a period of boiling-condensing mode operation in which a diluted volume of water is generated in the cold leg loop seals, followed by a reactor coolant pump start, the majority of flow from the active cold leg will impinge upon the core barrel and flow downward to the lower plenum. However, a fraction of this flow will travel radially around the downcomer and enter the non-active cold legs. One of the major resistances to flow in the alternate cold leg flow piping is that of the stalled RCP rotor. From previous works[28,30] the relative resistance of the core regions to that of the total flow path can be determined for both prototype and model.

Analysis shows that the relative fractional resistance of the UMCP Facility core to that of the one loop flow circuit is significantly greater than that of the prototype. On first observation this seems surprising given the resistance to flow of an actual nuclear power plant core. However, since the 'fractional' resistance is of interest this result becomes understandable. That is because the fractional resistance of the prototypical stalled RCP rotor is 1.7 times as large of that found at UMCP. To accommodate this facet of the design, allowance has been made to insert orifice plates at the UMCP RCP exit flange to increase the overall flowpath resistance thereby forcing the fractional resistance of the core region to be more closely matched between prototype and model. A re-evaluation of the importance of the alternate flowpath resistances will be conducted after initial test data has been obtained.

Auxiliary Systems

Six auxiliary support systems are employed in the UMCP Facility. They are (1) the High Pressure Safety Injection (HPSI) system, (2) the salt mixing system, (3) the deionized (DI) water system, (4) the salt recovery system, (5) the component cooling water system, and (6) the boil-off system. The first five systems are self-explanatory. The boil-off system allows control of the OTSG secondary side water level. Differential pressure cells provide input to control the operation of peristaltic feedwater pumps. High pressure injection can be supplied through the use of two positive displacement pumps to any/all of the four system cold legs.

INSTRUMENTATION

The Data Acquisition System (DAS) at the UMCP facility employs two Hewlett-Packard units operating under the HP-BASIC language. An IBM 386SX provides the operator interface and communication between the two units as well as storage of the raw data. A schematic of the UMCP data acquisition system is shown in Figure 6. The first system contains six, 24 channel multiplexer cards and one 20 channel card which allows higher voltage and current inputs. Due to its faster speed and separate CPU, all seven multiplexer cards, for a total of 164 channels, on this unit are used to measure temperatures. The second unit measures direct current (DC) voltages from system instruments including pressure transducers, differential pressure cells, and conductivity probes. It also allows for six remote control output signals.

Conductivity Probes

Conductivity is a measure of the ability of a material to conduct electrical current - or the inverse of resistivity. For aqueous salt-water solutions, ionic concentrations are directly related to the conductance of the fluid. The most common type of conductivity measuring instrument is that of two metal electrodes of known surface area which are separated by a known distance. Although this type of conductivity instrument is used frequently in industry it is unsuitable for use at the UMCP scaled model due to instrument size, range of measurement, and cost.

In the past, a unique design of conductivity measurement has been used for similar tests[32-34]. These instruments utilize a single relatively small metal electrode in contact with the fluid while the vessel container acts as a reference point. As described by Hemstrom[13], the major fraction of the electrical resistance is confined within a small radius around the sensing electrode.

Two types of single electrode conductivity probes have been installed in the UMCP facility. The 0-5 VDC output signal corresponds to the solution conductivity over the range of 0-500,000 μS/cm.

The first type of sensor is a glass bead with Kovar steel rod as shown in Figure 7. A hole is drilled directly through the bead and rod. The response time of this detector type depends upon how fast fluid is transported into the drilled hole of the bead. If turbulence is high and/or the direction of flow is into the hole then the response is relatively quick, being ~1 second and less. Temperature compensation is provided for by means of polynomial fits. Fourteen glass bead conductivity probes have been installed in a horizontal plane in the UMCP lower plenum. The probes are inserted through 3 existing instrumentation ports at an elevation just below the core barrel flow distributor. A plan view of these instruments is also given in Figure 7. Three more of the glass bead design conductivity probes have been installed along the flowpath of a single cold leg pipe. A final glass bead instrument has been installed in the downcomer of the vessel. This probe is situated at the elevation of the Reactor Vessel Vent Valves (RVVV). It should, in addition to providing fluid mixing information, act as a level indication since the measured signal from these instruments decreases sharply when uncovered.

The second type of conductivity instrument is a Kovar rod inside a Teflon casing. The response of this design is nearly instantaneous but unfortunately non-linear. The probe constant is very small and this instrument is best suited for tests involving large changes in solution concentrations. This approach is useful if one wishes to track the slug front as it passes through a region with little regard to the absolute value of actual conductivity. Four of the plain-tip design electrodes have been installed in the lower cold legs of the UMCP facility. Each is accompanied by a nearby thermocouple.

Since conductivity is a measure of the ionic mobility of a solution, it increases with increasing temperature. For aqueous salts this change due to temperature is typically in the range of 2%/°C. To account for this behavior a thermocouple is placed nearby to each conductivity measurement. In bench-top tests, the voltage response of the detectors is highly linear with respect to temperature at constant concentration. A typical probe response (glass bead) is shown in Figure 8.

Differential Pressure Transducers

Twenty-two differential pressure transducers are employed at the UMCP facility to provide indication of collapsed liquid levels in system components. Additionally, four higher range D/P cells are installed across each RCP inlet and outlet. In order to monitor the formation of boron-dilute volumes in the steam generators during BCM operation, a combination of D/P instruments are employed. The first D/P cell connects across the entire primary height (3.91 m) of the OTSG. The reference leg of this instrument is filled with unsalted (unborated) water. A second instrument is referenced to salted (borated) water and is connected across the primary loop seal elevation each cold leg (2.21 m). Stainless steel tubing (3/8") has been installed into the primary side tubing of each SG up to the loop seal elevation. The reference leg of this instrument remains salted throughout a test since the lowest level to which the SG primary liquid can fall is that of the cold leg loop seal. A schematic of this arrangement is shown in Figure 9. An equivalent height of liquid can be obtained from the measured pressure differences. For the instrument across the loop seal height

$$h_{LS}\rho_{sw} = h_{int}\rho_{sw} + (h_{LS} - h_{int})\rho_{sw} + \Delta P_{SR} \tag{2}$$

while the analogous equation for the D/P instrument across the entire SG primary side will be

$$h_{SG}\rho_{uw} = h_{int}\rho_{sw} + (h_T - h_{int})\rho_{uw} + \Delta P_{FR} \tag{3}$$

where 'uw' and 'sw' represent unsalted and salted water respectively. These two equations allow the evaluation of the interface position between the unsalted and salted water, h_{int}, and the total liquid water in the steam generator, h_T, given the measured differential pressures, ΔP_{SR} and ΔP_{FR}.

Temperature Measurements

The UMCP facility is instrumented with over 200 thermocouples of which 164 are continuously measured and stored. Two types of thermocouple (K,T) are currently in use at the facility. The thermocouples are heavily concentrated in the downcomer and lower plenum regions of the facility. A procedure which divides these regions into levels has been developed to allow identification of the various instruments. A total of twelve (12) levels have been identified in the downcomer and lower plenum as depicted in Figure 10. The lowest level is comprised of only 2 TC's located 1 inch above the hemispherical insert of the lower plenum. The highest level provides for temperatures in the annular downcomer above the elevation of the RVVVs. The rationale guiding the placement of the TCs in the downcomer is based upon the circumstance that for gravity driven mixing transients the region of interest will be the downcomer below the cold leg entrances. In addition, the TCs were aligned above one another so that temperatures could be employed to follow the mixing process (i.e. to track a colder, more dense slug of water). Exact cylindrical coordinates of each TC location have been documented in the report by Green.[35]

The remaining thermocouples have been placed in the facility piping and pressurizer. The numerous instrumentation ports in the cold and hot leg piping allows for convenient interchange of thermocouples depending upon the area(s) of interest. Currently, cold leg A1 has been more heavily instrumented.

Pressure

Primary system pressure is measured at three locations; the pressurizer, the hot leg at the candy cane, and the cold leg. The pressure transducer attached to the pressure vessel is used to provide a continuous pressure signal to a strip-chart recorder.

Core Power

The redesigned UMCP Facility is capable of providing 115 kW of power using nineteen (19) cartridge heaters. The use of two heater types allows for the possibility of marginal core uncovery without severe consequences. The nine (9) 3 kW heaters are designed to operate in gas or liquid while the ten (10) 8.8 kW heaters must operate while immersed in liquid. The heaters used for the experiment are 3 phase, 480 volts AC, delta connected, resistance heaters. Heater power is supplied using Silicon Controlled Rectifiers (SCR) under the control of the DAS.

The 8.8 kW heaters are 1" diam x 44" long, incoloy sheathed cartridges with a heated length of 14" below an unheated length of 24". An internal type K thermocouple in each heater is used to determine heater temperature so that burnout can be avoided. These heaters must be covered by water during operation. The 3 kW heaters are 1" diam x 51" long, incoloy sheathed cartridges with a heated length of 24" below an unheated length of 26". Both high and low power heaters begin at the lower core barrel plate, therefore the heated section of the 3 kW heaters extends 10" above that of the higher power heaters. Figure 11 shows a schematic of the heaters and their arrangement. The heaters have been divided into four banks for the purpose of power control. Power generation is not symmetric yet this is of little consequence to the test program. Independent measurements (resistance and voltage) allow for a direct calculation of the heater power. A feedback system is in place and calibrations have been conducted such that core power is remotely controlled from the 386 PC.

Reactor Coolant Pumps

The UMCP facility has four centrifugal reactor coolant pumps (bottom suction, side discharge) in each cold leg. The pumps were scaled according to the original Facility design and no modifications were made to support boron-mixing experiments. The 3HP motor speed of each pump can be controlled within the range of frequencies from 0 to 80 Hz (corresponding to 0 to 1500 rpm) using the remote DAS. Nominal steady state operation at 1160 rpm for water at 5 bar / 120°C yields 60 gpm equating to a cold leg velocity of roughly 1 m/s.

TEST MATRIX & INITIAL RESULTS

The test matrix identifies numerous tests distributed over three tasks. The three tasks are: a) establishment of a boron-free water volume under SB-LOCA conditions; b) transport of boron-free water and mixing mechanism; and c) testing associated with ex-vessel transport and mixing.

The mapping of the operational conditions for which the system transfers heat from the reactor vessel to the steam generators in the Boiler Condenser Mode (BCM) and in the Interruption Resumption Mode (IRM) is the objective of early tests. The quantification of the heat losses in a long term test will also be conducted. Of interest is the new heater rod configuration and the fact that the facility does not currently use insulation.

The BCM is established when and adequate condensing surface is present in the steam generators and yet the natural circulation of liquid above the candy canes is interrupted. A condensing surface is exposed when the collapsed liquid interface of the primary system is lower than the secondary liquid level. This heat transfer mode requires a system inventory which is low enough to achieve the desired condensing surface in the steam generator. Therefore, the maximum system inventory is bounded by the secondary inventory. Furthermore, it is also necessary to have a sufficiently low inventory in the primary to inhibit the flow of liquid above the candy canes. Which of these two conditions is limiting depends on the secondary level set point. For high values of the secondary level set points, the non liquid flow condition is crucial while for low secondary level set points, the existence of condensing surface becomes paramount. Such mapping tests proceed through a series of steady-states conditions achieved with the maximum primary inventory for various secondary set points. Note that as the condensing surface is diminished, the system operating pressure increases.

Heat transfer during IRM is rather complex and the starting conditions as well as the propagating nature of this intermittent process will be mapped. Traditionally we have always approached the IRM while draining the system in an SB-LOCA type transient. We will extend the IRM initiating conditions to the case where either the primary inventory is increased or the secondary is lowered while the primary is in the proper inventory range. Inspection of the previous database on SB-LOCA as well as the current experimentation will identify the required conditions for the establishment and sustainability of IRM.

Upon completion of these initial mapping tests, segregation of boron-dilute volumes under a range of conditions will be performed. These tests center about a nominal test which will be performed at a power of 90 kW, with an intermediate secondary liquid level set point of ~75%. Two variations on secondary elevation and core power respectively are planned.

Having established a segregation mechanism the test matrix will proceed to study the transport of fresh water slugs towards the core. Two classes of phenomena are anticipated: a) controlled transport and b) natural transport. Controlled transport indicates those actions such as HPI or RCP activations which will move boron-free water to the core. More complex interactions whereby the system changes its energy transport mode and thus causes the displacement of boron-free water from its segregated location fall under the second category.

It is realized that as the program develops these initial plans might be changed to accommodate more pressing issues. As of this writing two issues which may require further testing are: a) additional natural transport scenarios and b) ex-vessel mixing phenomena.

The ex-vessel mixing issue is important when considering local mixing events in the path from the steam generators to the vessel. The cold leg geometry allows for a long vertical mixing zone, for an horizontal stratification zone and for local mixing zones such as the HPI injection location. Based on the overall system response from the previous tests, it is conceivable that specific mixing issues in one or more of these zone might require some detailed testing if the mixing turns out to be significant in the overall process. As a possible example, consider that there might be a buoyant effect in the vertical cold leg riser as the boron-free water starts entering the bottom of the cold leg. Similarly consider the potential for segregated/mixed stratified flow as the water flows over the loop seal and down toward the vessel. It is possible that for some of these tests, the conductivity probes could be concentrated on one cold leg to allow for the required spatial resolution.

Tests have been included which assess the repeatability of the data. These tests should allow for ample characterization of the consistency of the experimental database. Note that the integral nature of the facility, requires this kind of investigation due to the sensitivity of the system to the feedback and interactions that are present in most of these transients. The multi-loop, multi-components integral system is expected to exhibit under certain circumstances some degree of stochastic behavior even if the transient trajectory is observed to transit though a number of common system states.

SELECTED TEST RESULTS

Several tests from the projected test matrix have been completed at the UMCP Facility. Table 4 lists the test name and selected features of the test. Note that each test is designated by the date at which it was conducted. Three specific tests will be examined here in more detail;

BOR080295 - System operational regime test
BOR080495 - Steady state SNC energy balance test
BOR082395 - Salt-free volume generation test

System Operational Regime Characterization Test - BOR080295

The purpose of this test was to map the various operational regimes under which the UMCP Facility operates. From the initial nominal conditions of 90 kW and 70% SG secondary level the facility progressed through single phase natural circulation (SNC) through and extended period of IRM operation followed by complete flow interruption and extended operation in BCM. The relationship between system inventory and operating regime was clearly demonstrated as the HPI system (through only a single cold leg) was used to restore system inventory and to restore the facility to an IRM operation. Continued filling of the system using HPI restored the facility to a single phase condition. It was noted that since the HPI fluid is not initially de-gassed, that its introduction into the hot system results in the addition of noncondensable gases. These gases were found to accumulate at the highest portions of the hot leg candy canes and inhibit the re-initiation of natural circulation single phase flow patterns. System pressure for this test is depicted in Figure 12 (a) with the operational regimes noted. Correspondingly, Figure 12 (b) shows the system pressure versus the system inventory. It is clear from this and other tests that the operational regimes are dependent upon the system inventory. Features of the IRM mode of operation are evident in Figure 13 such as the dramatic temperature and level changes induced by condensation in the cold legs.

Energy Balance Test - BOR080495

The objective of this test was to obtain an estimate of the heat losses from the system during single phase natural circulation (SNC). A nominal power of 90 kW was requested from the DAS and local measurements showed an average power of 90.4 kW.

Pressurizer heaters were cycled to maintain a nearly constant ($\pm$1 psig) system pressure of 65 psig. The SG secondary water level was maintained at 90%.

Two independent methods which made use of instrumentation in the SG feedwater system, were employed to determine energy removal rates from the UMCP steam generators of 81.9 and 82.3 kW. It was evident that there is a slight asymmetry in the energy removal with the B-side SG being slightly more effective (44 vs 38.3 kW)..

To asses the energy losses from the system metal surfaces to the ambient, fifteen J-type surface thermocouples were distributed on the major heat loss surfaces of the facility. Using standard convection correlations and the known heat transfer areas it was determined that 7.8 kW of the heater energy input was lost to the ambient environment at steady state conditions. Even if one considers that during BCM operation the upper portions of the facility will be steam-filled and therefore at higher temperatures, the energy losses to the environment are only slightly changed.

Salt-free Volume Generation Test - BOR082395

In this test the nominal conditions of 90 kW heater power and 75% SG secondary water level were maintained. A total of 290 liters of the initially uniform10% $NaNO_3$ primary fluid was removed and the system placed into a steady BCM operation. The break size was increased for this test (from 1/16" to ½") in order to quickly pass through the IRM phase. This type of operation proceeded for ~30 minutes at which time the test was ended. A drain arrangement at the bottom of each SG allowed for collection of the salt-free volume which was generated. Samples of the drained fluid were taken and evaporated to determine the salt concentration. The results of these measurements are seen in Figure 14. The figure shows clearly that salt-dilute fluid has accumulated in the SG. The test also provided an opportunity to check the usefulness of the combined D/P cell configuration for calculating the salted-unsalted water interface in the OTSGs. Verification of this method was obtained from the collected drain samples. Selected parameters from this test are reproduced in Figure 15.

The response of selected conductivity probes in the lower plenum are shown for a recent test during extended BCM operation (BOR090895) in Figure 16. From ~1800 seconds onwards the associated temperatures at these locations are constant. This shows the probes ability to measure the increase in concentration of the salt solution in the core region as a result of boiling. The quick drop at the end is due to the starting of a RCP. Unfortunately, during this particular test the UMCP DAS records data at ~4 second intervals and this is far too slow to track any mixing of the salt-dilute slug. Increased data acquisition speed will be provided for in the transport and mixing phases of the project.

SUMMARY

The UMCP 2x4 Facility is being employed to study the generation, transport and mixing of boron-dilute water volumes. The scaling rationale and facility modifications have been presented along with the instrumentation layout and proposed test matrix. Initial test results show that boron-dilute water volumes can be produced in the OTSG and cold leg loop seals of the facility as a result of boiling-condensing mode operation after a simulated SB-LOCA. Upcoming tests will focus upon the transport and mixing of the boron-dilute water volume in the downcomer and lower plenum of the test facility.

TABLE 4 - TESTS COMPLETED at UMCP 2x4 LOOP

TEST	Power	SG Level	Fluid	Comment
BOR062795	n/a	n/a	DI	Hydrostatic Pressure test
BOR062996	varied	varied	DI	Steaming Tests
BOR070695	90	75	DI	Blowdown w/ PZR
BOR070795	90	75	DI	Intermittent Blowdown
BOR071495	90	75	DI	System Testing
BOR072195	90	70	DI	290 liters removed
BOR072695	90	90	DI	300 liters removed
BOR072895	90	75	DI	Blowdown & HPI Refill - 1 leg
BOR080295	90	75	DI	Blowdown & HPI Refill - 1 leg
BOR080495	90	90	DI	Energy Balance - SNC
BOR080995	90	50	5% $NaNO_3$	Heater Scram
BOR081895	90	75	DI	Blowdown & HPI Refill (all)
BOR082395	90	75	10% $NaNO_3$	BCM operation for 1 hr
BOR082595	90	75	10% $NaNO_3$	repeat of 0823
BOR083095	90	75	10% $NaNO_3$	BCM & RCP test
BOR090195	90	see note	10% $NaNO_3$	SG level control problems
BOR090895	115	75	10% $NaNO_3$	BCM operation for 45 min.
BOR091595	75	75	10% $NaNO_3$	BCM operation for 2 hrs
BOR092395	90	90	10% $NaNO_3$	BCM operation for 1 hr

REFERENCES

1 Hyvärinen, Juhani, *The Inherent Boron Dilution Mechanism in Pressurized Water Reactors*, Nuclear Engineering and Design, **Vol. 145**, pp. 227-240, (1993).

2 Diamond, D. J., P. Kohut, H. Nourbakhsh, K. Valtonen, and P. Secker, Probability and Consequences of a Rapid Boron Dilution in a PWR, NUREG/CR-5819 (BNL-NUREG-52313), (1992).

3 Jacobson, S., *Risk Evaluataion of Local Dilution Transients in a Pressurized Water Reactor*, Ph.D. Dissertation - Department of Mechanical Engineering, Linköping University, No. 275, 1992.

4 Macian-Juan, Rafael, A Preliminary Study on the TRAC-BF1/MOD1 Simulation of Boron Injection Following an ATWS, Nuclear Engineering Department - Pennsylvania State University, (1994).

5 Sun, J. G. and W. T. Sha, Analysis of Boron Dilution in a Four-Loop PWR, NUREG/CR-6266, ANL-94/35, (March, 1995)

6 Sun, J. G. and W. T. Sha, Analysis of Thermal Mixing and Boron Dilution in a PWR, NUREG/CR-5822, ANL-91/43, (February, 1993).

7 San Onofre Boron Dilution Scenario - June 06, 1993

8 di Marzo, M., K. Almenas, Y. Hsu, and Z. Wang, *The Phenomenology of a Small Break LOCA in a Complex Thermal Hydraulic Loop*, Nuclear Engineering and Design, **Vol. 110**, pp. 107-116, (1988).

9 Hsu, Y. Y., Z. Y. Wang, C. Unal, M. di Marzo, and K. Almenas, *Scaling-Modeling for Small Break LOCA Test Facilities*, Nuclear Engineering & Design, Vol. **122**, pp. 175-194, (1990).

10 Wang, Z. W., C. Unal, M. di Marzo, Y. Y. Hsu, and K. Almenas, *Use of Inventory as a Chronological Scale of Small Break LOCA in the IST Facilities*, International Journal of Heat & Technology, Vol. **7**, No. 3-4, pp. 36-44, (1989).

11 di Marzo, M., K. Almenas, and S. Gopalnarayanan, *Core to Surge-Line Energy Transport in a Severe Accident Scenario*, Nuclear Engineering & Design, **Vol. 154**, pp. 119-132, (1995).

12 Tinoco, H., B. Hemström, and N. Andersson, Physical Modelling of a Rapid Boron Dilution Transient, VS-S 93:B21, Vattenfall AB, Sweden, (September, 1993).

13 Hemstrom, B. and N. Andersson, Physical Modelling of a Rapid Boron Dilution Transient -The EdF Case, VU-S 94:B16, Technical Report Vattenfall Utveckling AB, Sweden, (July, 1994).

14 Chen, C. J. and C. P. Nikitopoulos, *On the Near Field Characterization of Axisymmetric Turbulent Bouyant Jets*, International Journal of Heat and Mass Transfer, **Vol. 22**, pp. 245-255, (1979).

15 Henderson-Sellers, B., Modeling of Plume Rise and Dispersion - The University of Salford Model: USPR, Lecture Notes in Engineering, Edited by C. Brebbia and S. Orszag, Springer-Verlag, Berlin, (1987).

16 Iyer, K. and T. G. Theofanous, Flooding-Limited Thermal Mixing: The Case of High Froude Number Injection, Proceedings of the Third International Topical Meeting on Reactor Thermal Hydraulics, Newport, RI, pp. 13.E-1 - 13.E-6, (October, 1985).

17 Theofanous, T. G., H. P. Nourbakhsh, P. Gherson and K. Iyer, Decay of Buoyancy Driven Stratified Layers with Applications to Pressurized Thermal Shock (PTS), NUREG/CR-3700, (1984).

18 Theofanous, T. G. and K. Iyer, *Mixing Phenomena of Interest to SBLOCAs,* Nuclear Engineering and Design, **Vol. 102**, pp. 91-103, (1987).

19 Yoshimura, S., T. Emori, T. Fujimoto, A. Tsuge, and K. Koyama, Mixing Behavior of Safety Injection Water Within the Cold Leg and the Downcomer Under Postulated Accidents, Proceedings of the Third International Topical Meeting on Reactor Thermal Hydraulics, Newport, RI, pp. 13.D-1 - 13.D-8, (October, 1985).

20 Hassan, Y. A., A Three Dimensional Thermal and Fluid Mixing Analysis Using the Mass-Flow-Weighted Skew-Upwind Differencing Scheme, EPRI NP-3870, (February, 1985).

21 Theofanous, T. G., J. L. LaChance and K. A. Williams, *The Thermal Hydraulics of Small-Break Loss-of-Coolant Accidents Relative to Pressurized Thermal Shock, Nuclear Science and Engineering*, **Vol. 102**, pp. 74-100, (1989).

22 Rothe, P. H. and Fanning, M. W.,, Transient Cooldown in a Model Cold Leg and Downcomer, EPRI-NP-3118, (May, 1983).

23 Tuomisto, H. and P. Mustonen, Thermal Mixing Tests in a Semiannular Dowmcomer with Interacting Flows from Cold Legs, NUREG/IA-0004, (October, 1986).

24 Theofanous, T. G. and H. Yan, A Unified Interpretation of One-Fifth to Full Scale Thermal Mixing Experiments Related to Pressurized Thermal Schock, NUREG/CR-5677, (April, 1991).

25 Nourbakhsh, H.P. and Z. Cheng, *Potential for Boron Dilution During Small Break LOCAs in PWRs*, Technical Report W-6156 - Brookhaven National Laboratory, October, (1994).

26 Bayless, P. D., Analyses of Natural Circulation During a Surry Station Blackout Using SCDAP/RELAP5, NUREG/CR-5214, EGG-2547, (1988).

27 Popp, M., *The Simulation of Thermohydraulic Phenomena in a Pressurized Water Reactor Primary Loop*, NUREG/CR-4789, (1987).

28 Larson, T.K., An Investigation of Ingtegral Facility Scaling and Data Relation Methods (Integral System Test Program), NUREG/CR-4531, EGG-2440, (February, 1987).

29 Massoud, M., An Analytical and Experimental Investigation of Natural Circulation Transients in a Model Pressurized Water Reactor, NUREG/CR-4788, (January, 1987).

30 di Marzo, M., et. al., University of Maryland at College Park (UMCP) 2x4 Loop Test Facility, NUREG/CR-4843, Annual Report for 1985.

31 Floyd, J., J. Green, K. Almenas, and M. di Marzo, Generation, Transport and Mixing of Boron-Dilute Water in a Scaled Integral Model of a B&W Reactor, Submitted to the OECD/CSNI Specialists Meeting on Boron Dilution Reactivity Transients, State College, PA (October, 1995).

32 Chua, S. K., J. W. Cleaver, and A. Millward, *The Measurement of Salt Concentration in a Plume Using a Conductivity Probe*, Journal of Hydraulic Research, **Vol. 25**, No. 3, pp. 171-178, (November, 1986).

33 Gibson, C. H. and W. H. Schwarz, *Detection of Conductivity Fluctuations in a Turbulent Flow Field*, Journal of Fluid Mechanics, **Vol. 16**, pp. 357-364, (1963).

34 Andersson, N. G., Vidareutveckling av Konduktivitetsmatnignsutrustning. Teoretisk Genomgang av Matmetodik, Report No. VU-S 94:B12, Vattenfall Utveckling AB, Sweden, (1994).

36 Alvarez, D., O. Daubert, L. Janvier, J.P. Schneider, Three Dimensional Calculations and Experimental Investigations of the Primary Coolant Flow in a 900 MW PWR Vessel, Proceedings of NURETH-5, Salt Lake City, (1993).

37 Alvarez, D., A. Martin, and J. P. Schneider, Boron Mixing Transients in a 900 MW PWR Vessel for a Reactor Startup-Up Operation, Proceedings of NURETH-5, Salt Lake City, (1993).

38 Gango, P., Numerical boron Mixing Studies for Loviisa Nuclear Power Plant, Proceedings of NURETH-7, NUREG/CP-0142, pp. 3096-3119, (September, 1995).

This work is being performed under USNRC contract NRC-04-94052

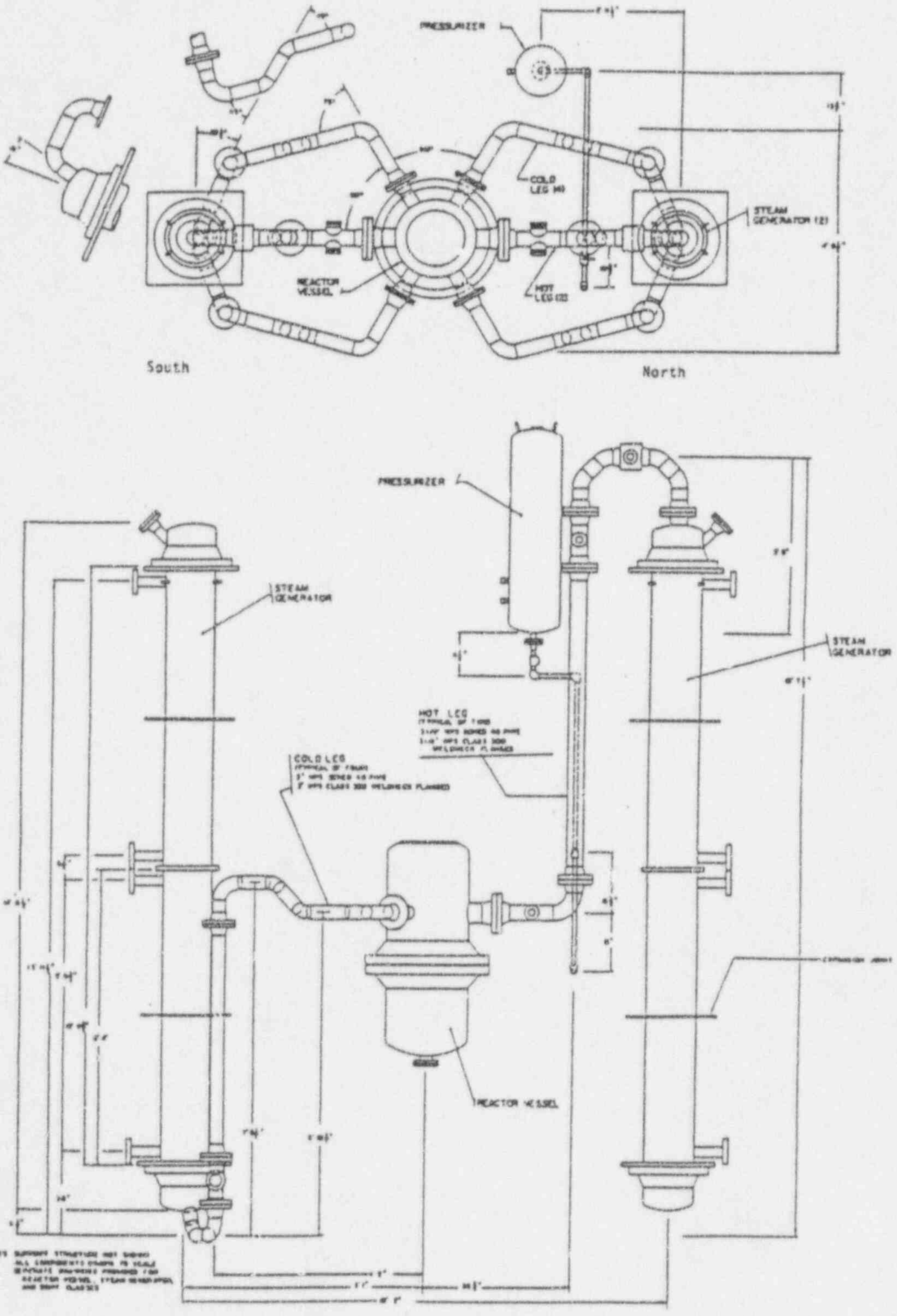


Figure 1 Schematic of UMCP 2x4 Loop Facility

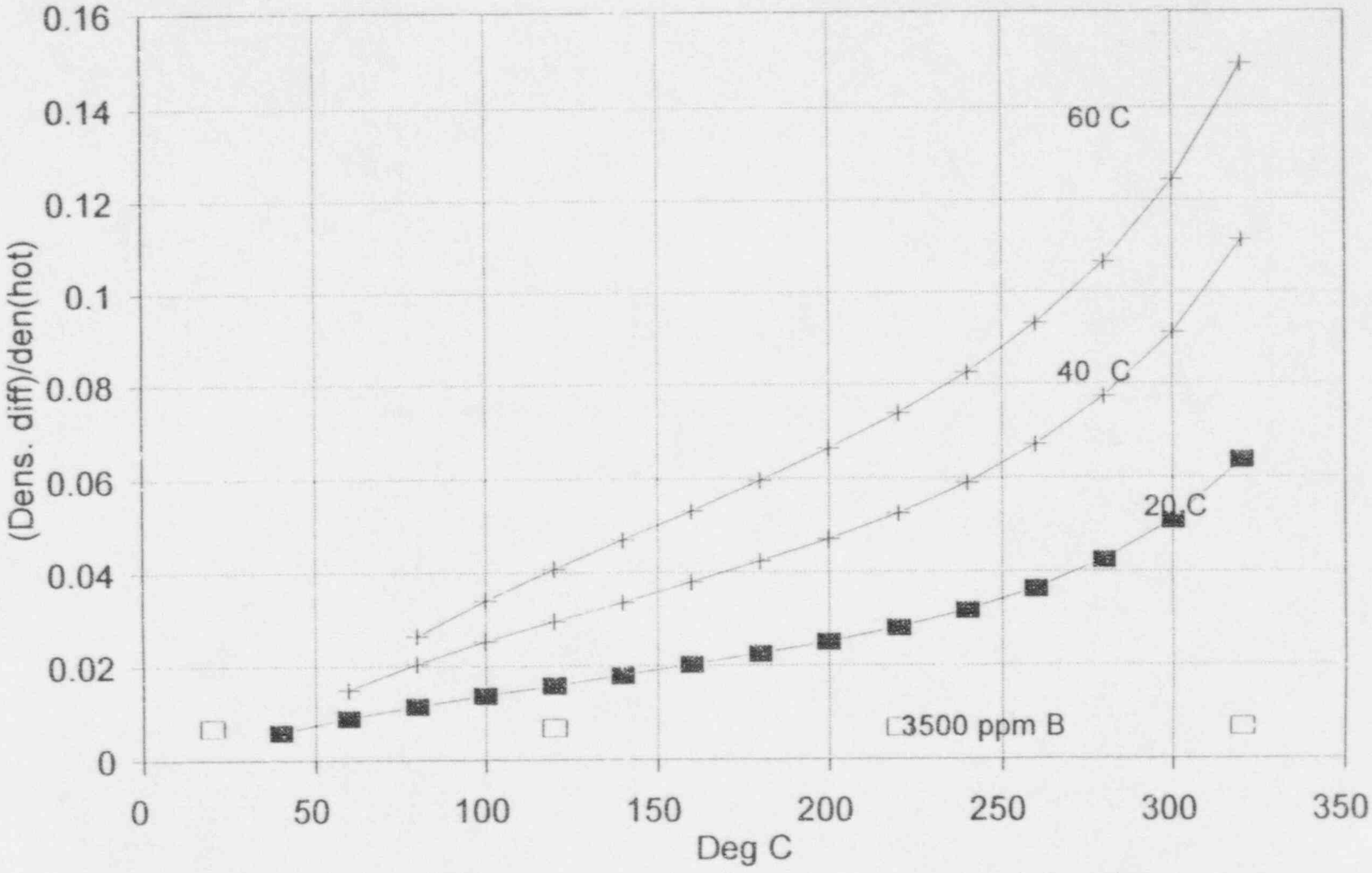
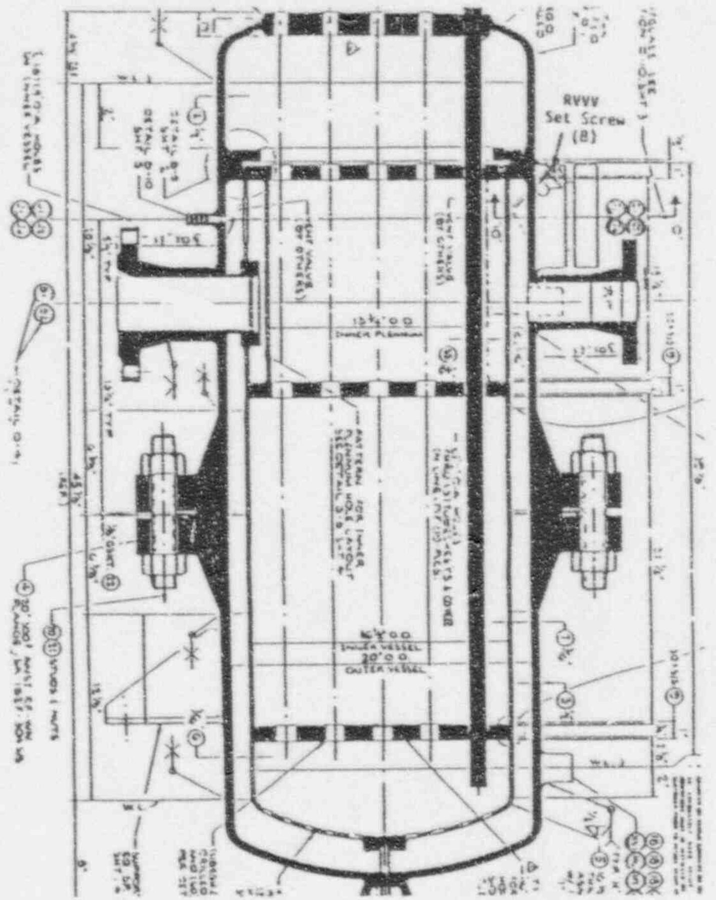


Figure 2 Effect of Temperature & Concentration of Density Ratios



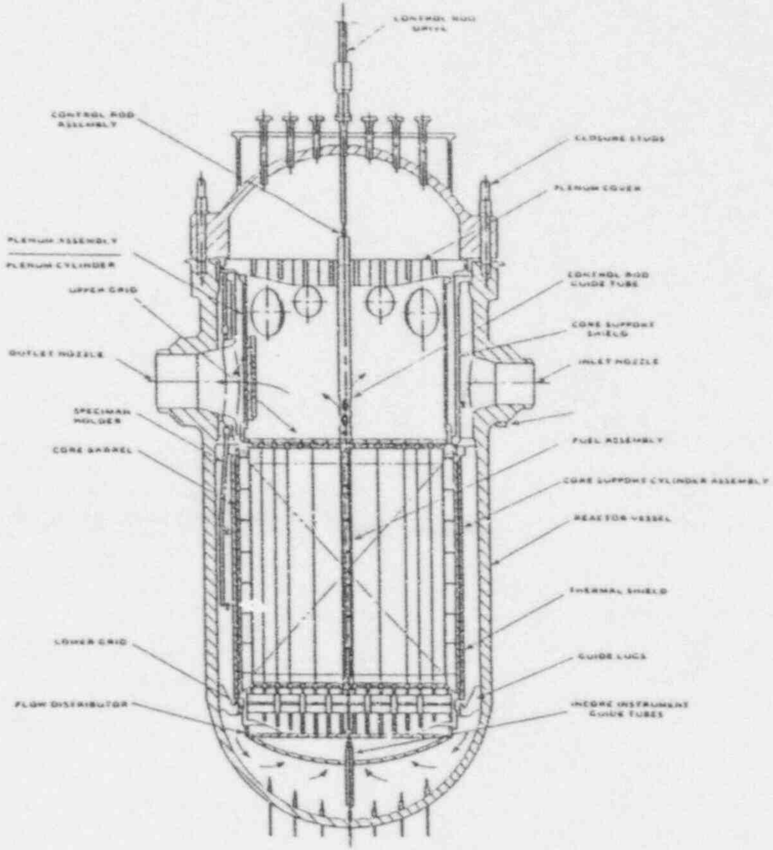


Figure 3 Model (pre-modification) and Prototype RPV

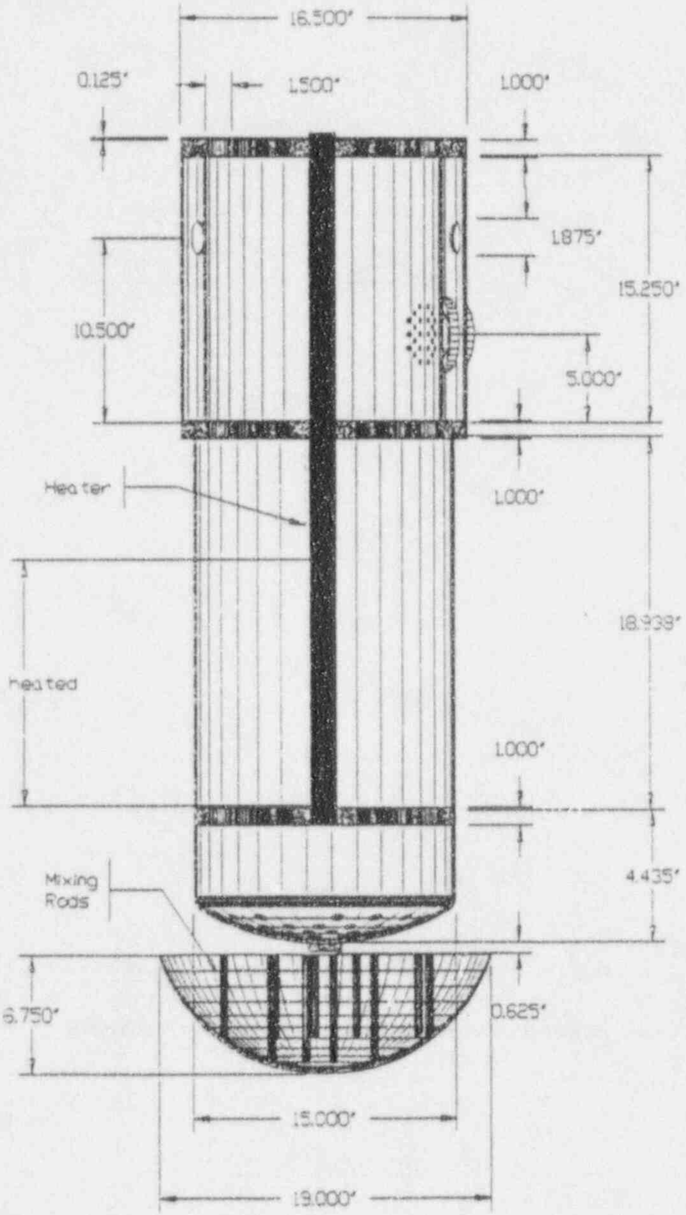
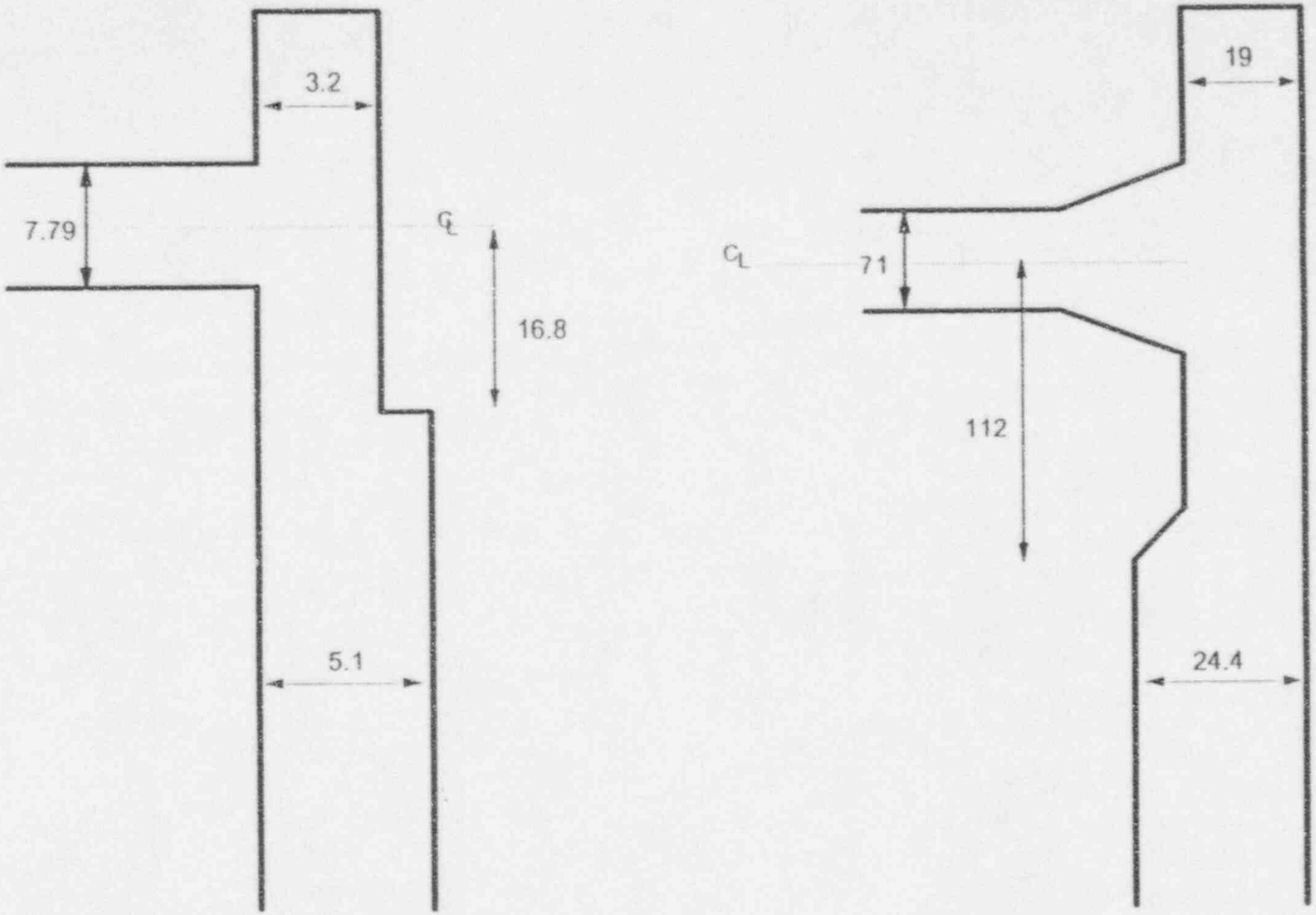


Figure 4 Modified UMCP Core Barrel & Lower Plenum Insert



(a) MODEL - UMCP

(b) PROTOTYPE - TMI

Mixing in Annular Downcomer Region

(not to scale - dimensions in cm)

Figure 5 Model and Prototype Downcomer Geometry

DATA ACQUISITION SYSTEM

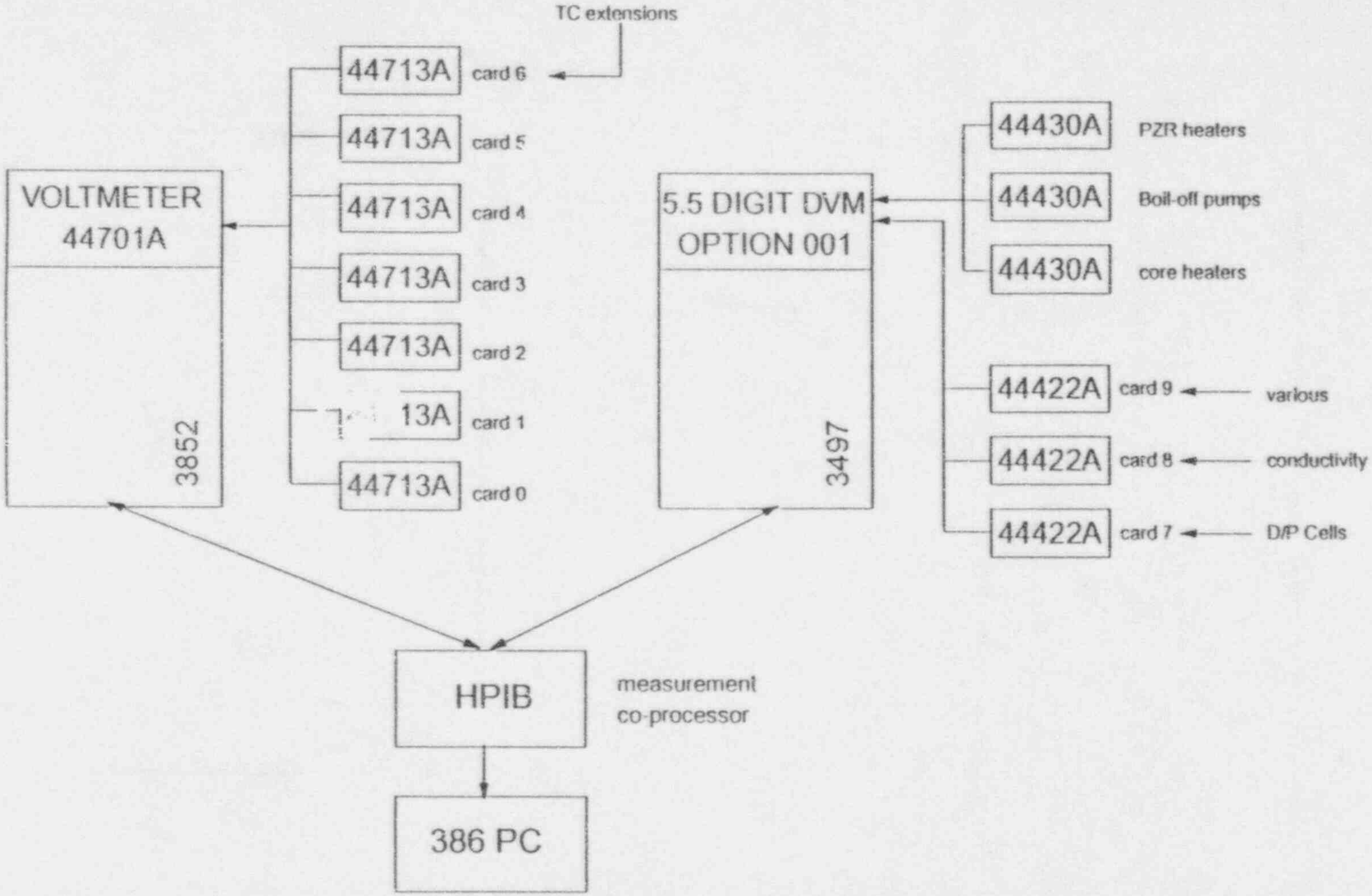


Figure 6 UMCP Data Acquisition System

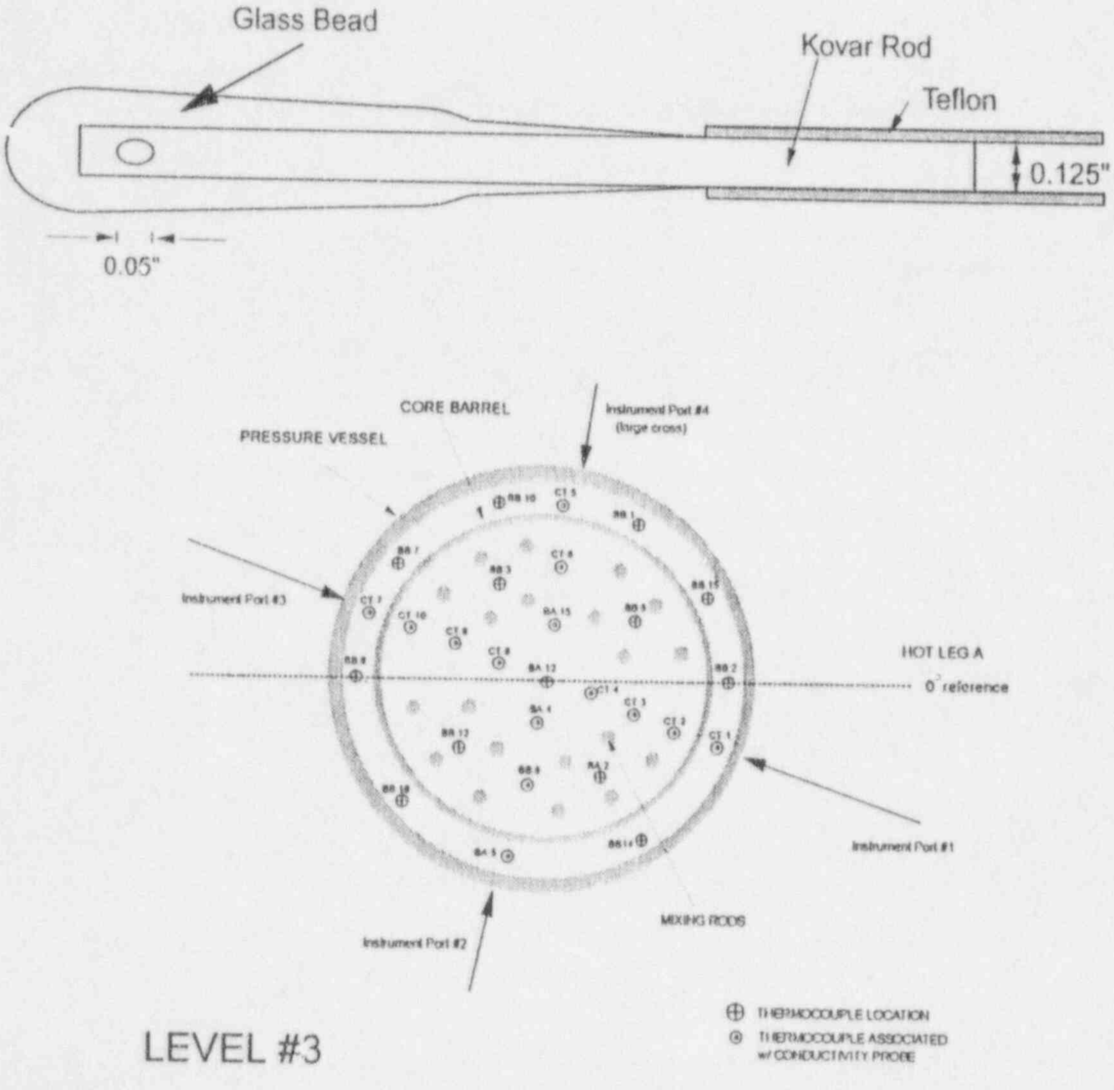


Figure 7 Conductivity Probe Design & Lower Plenum Arrangment

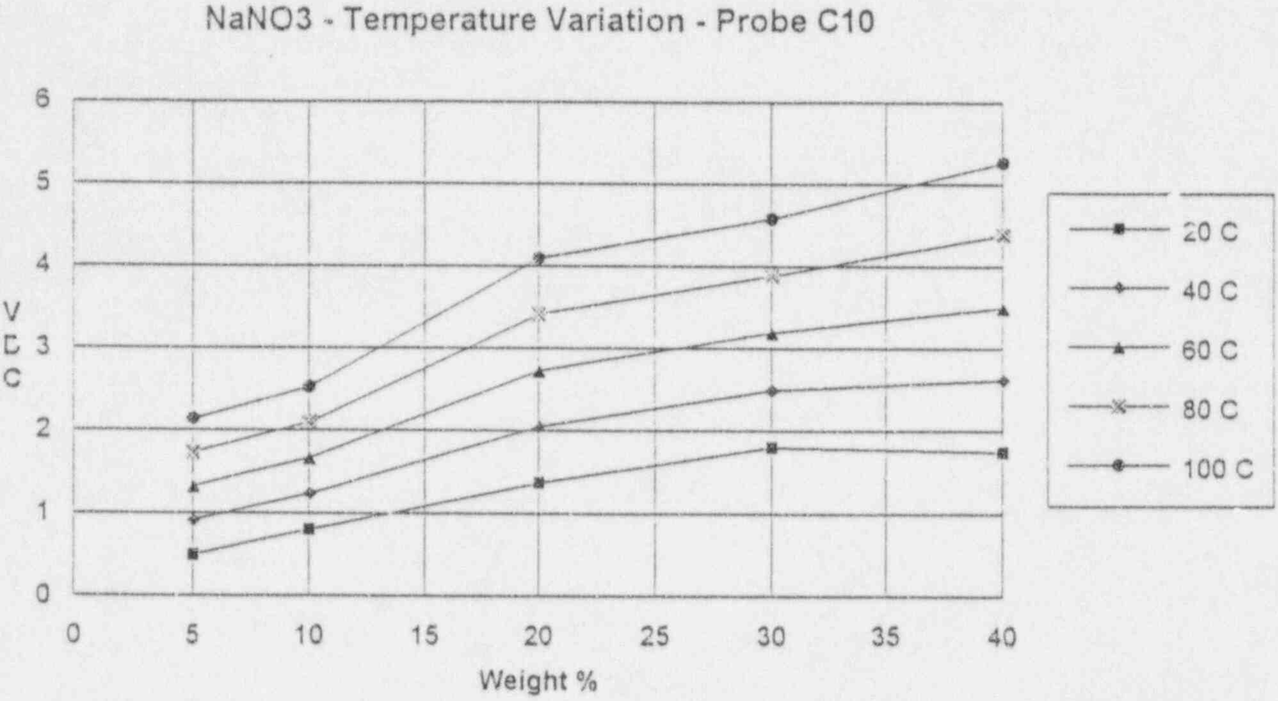


Figure 8 Conductivity Probe Response

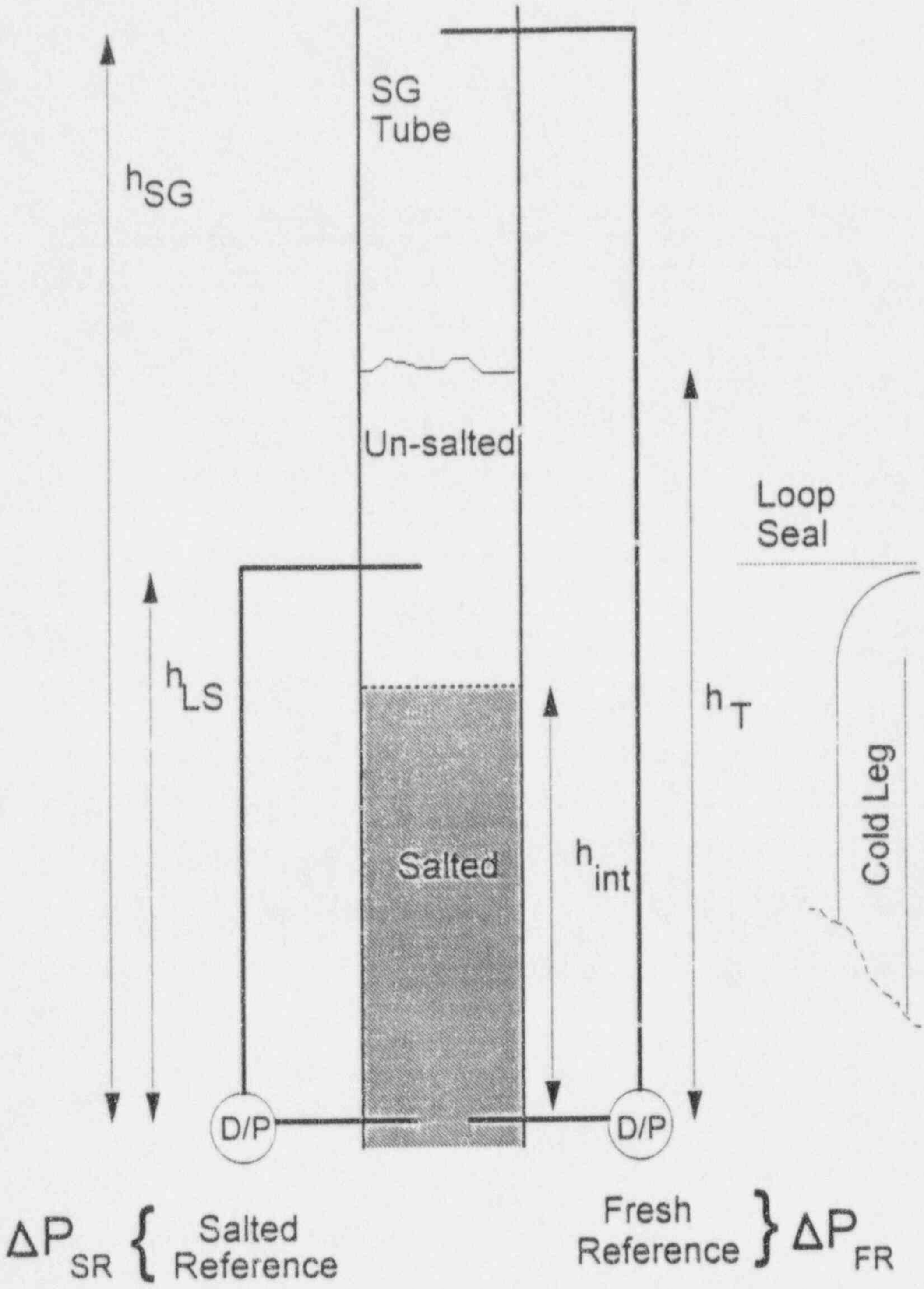


Figure 9 D/P Cell Arrangement in UMCP OTSG

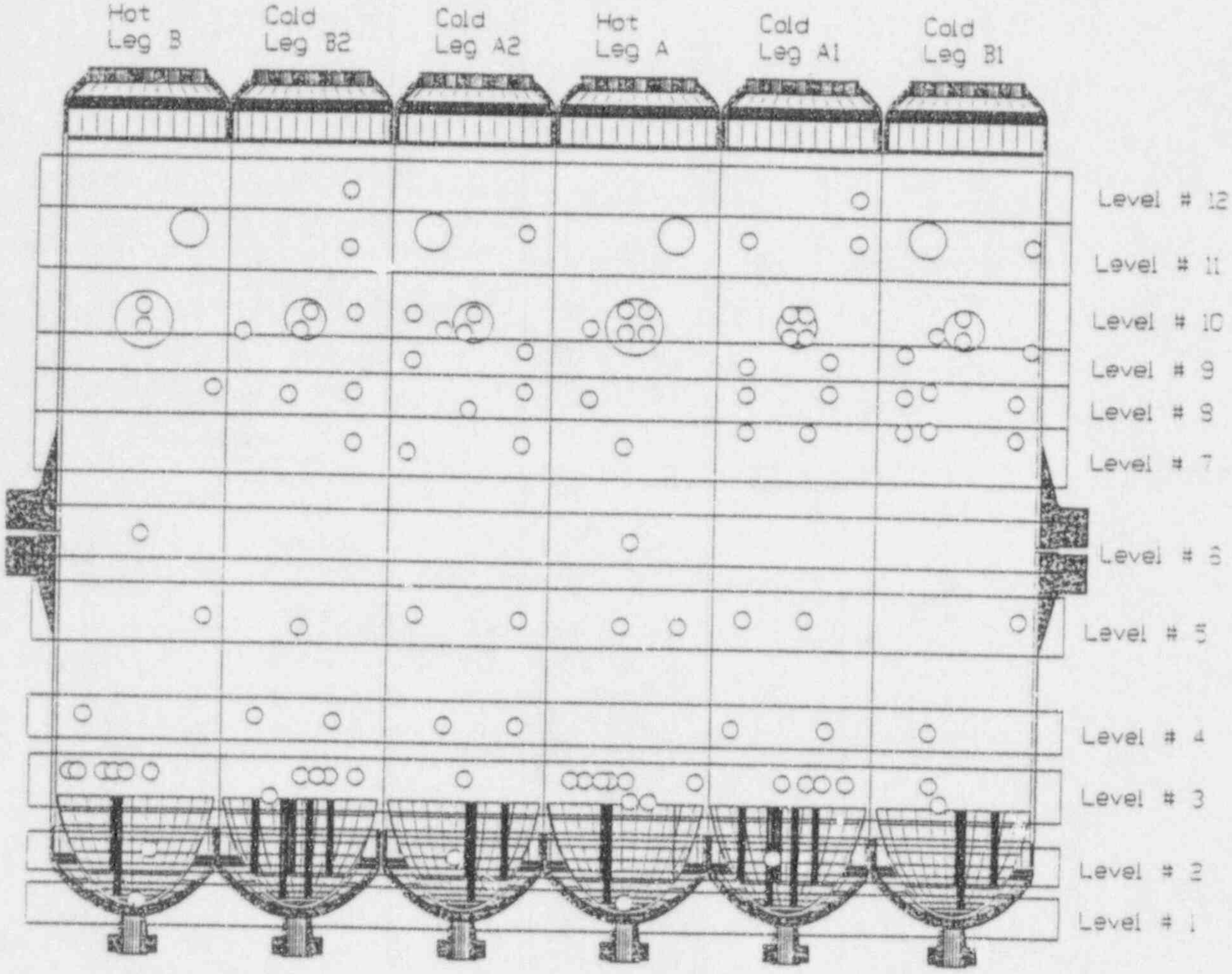


Figure 10 Thermocouple in UMCP Downcomer and Lower Plenum

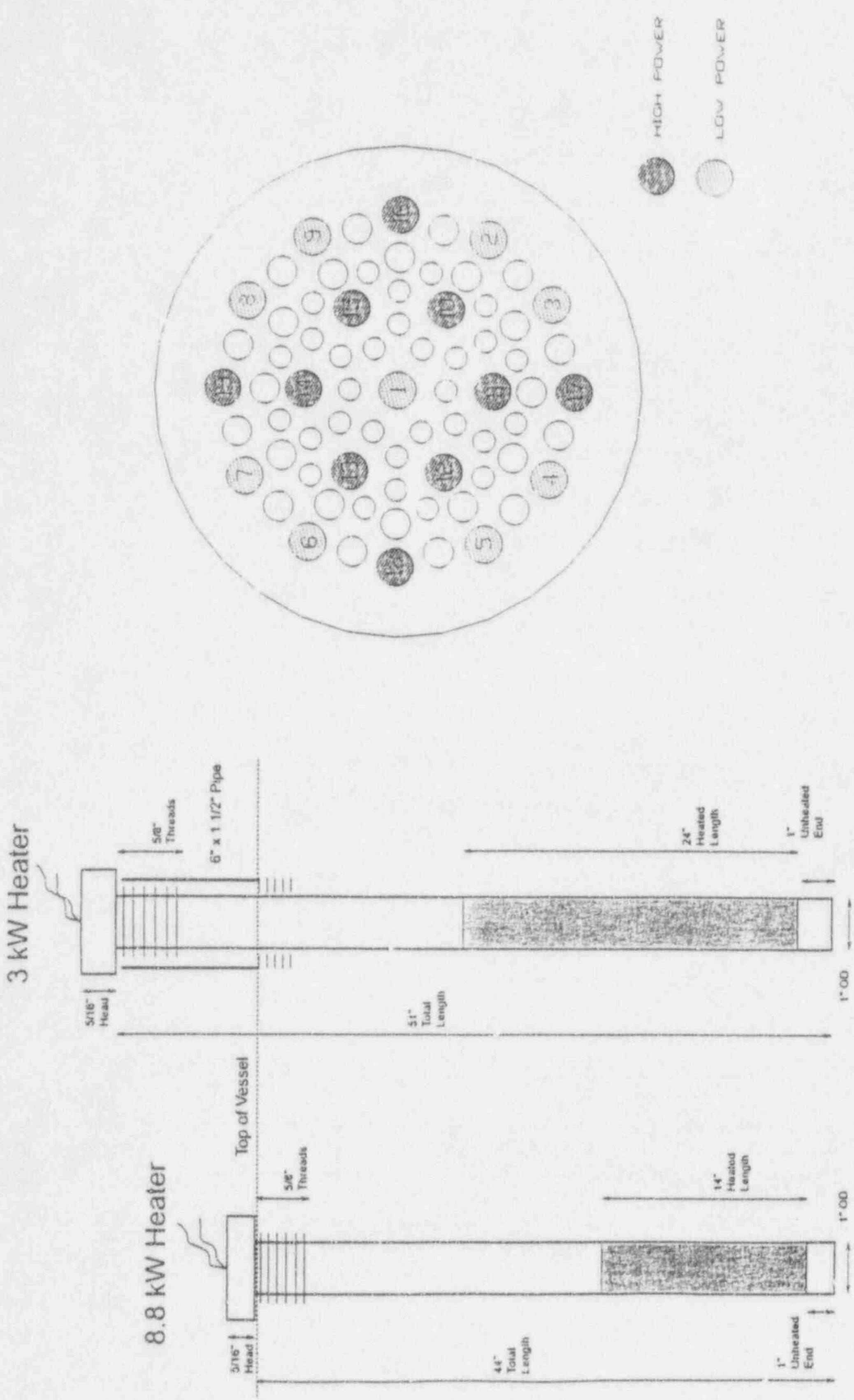


Figure 11 UMCP Heater Design

Figure 12 (a) - BOR0802_95
System Pressure

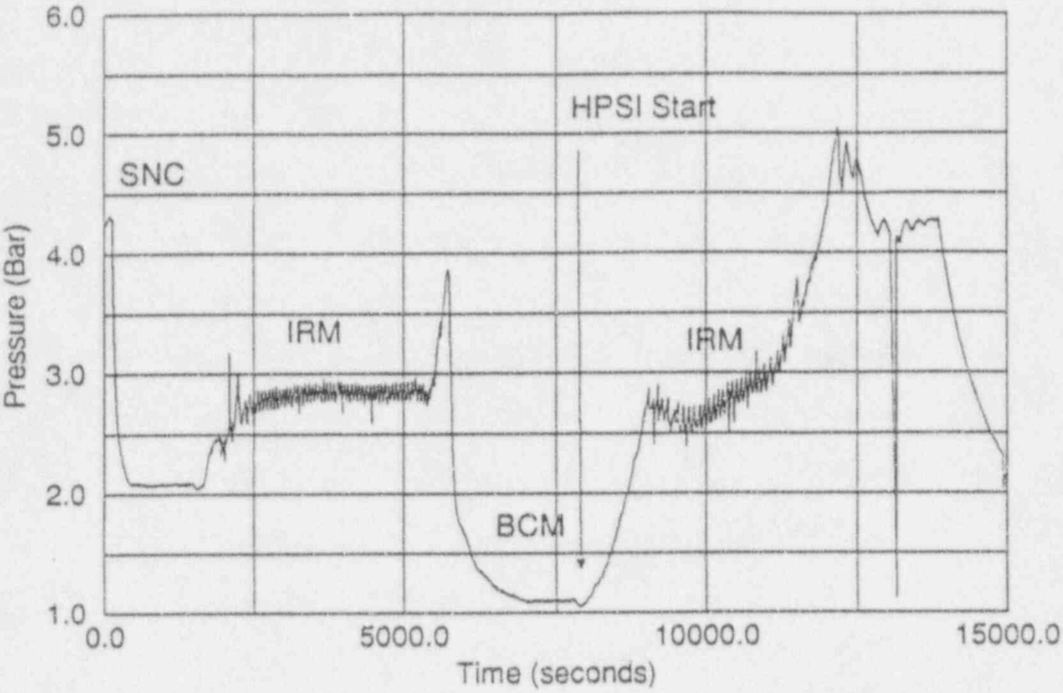
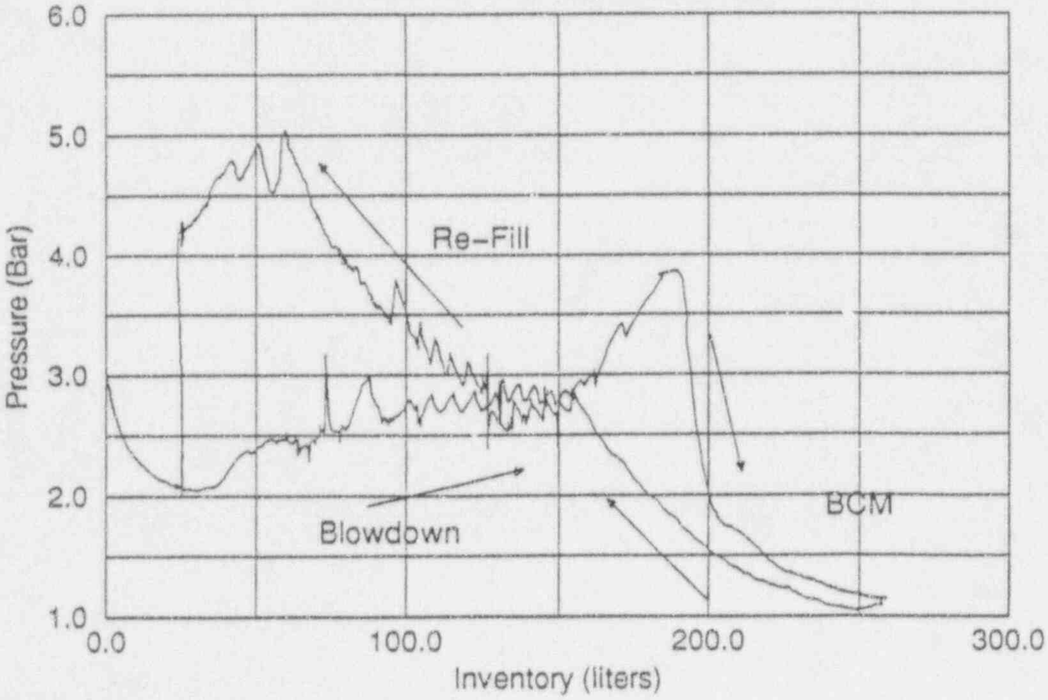


Figure 12 (b) - BOR0802_95
Pressure - Inventory Relationship



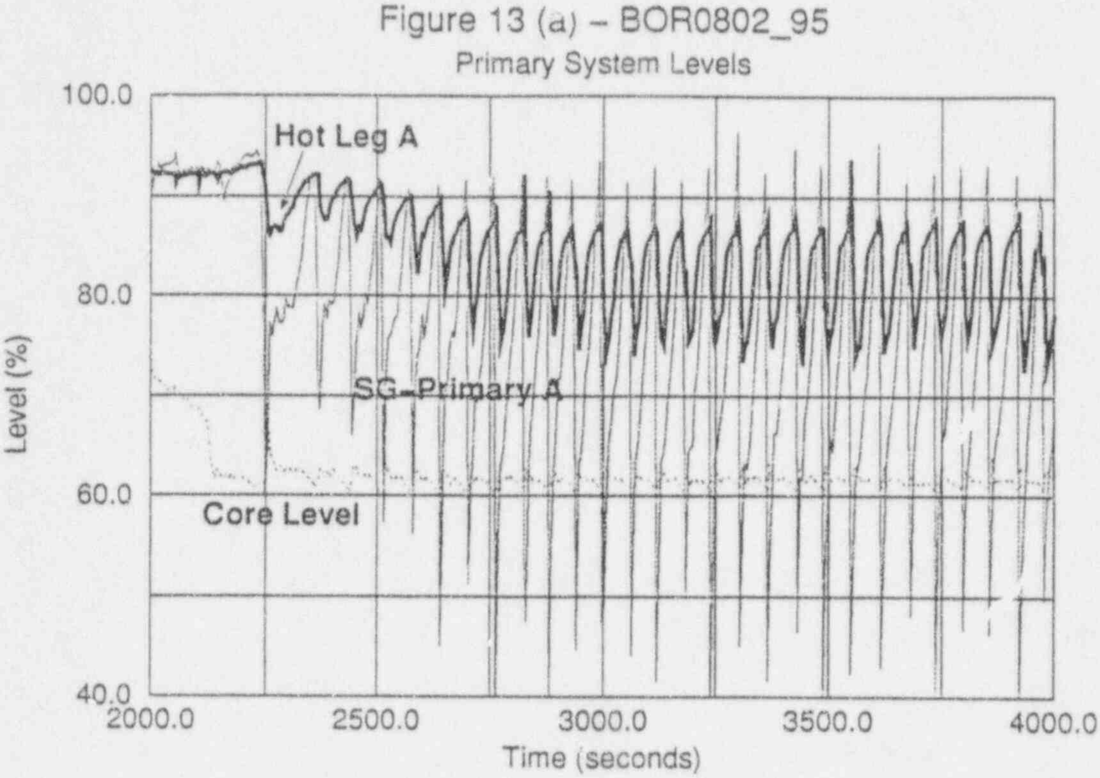


Figure 13 (a) - BOR0802_95
Primary System Levels
100.0
80.0
60.0
40.0
Level (%)
Hot Leg A
SG-Primary A
Core Level
2000.0
2500.0
3000.0
3500.0
4000.0
Time (seconds)

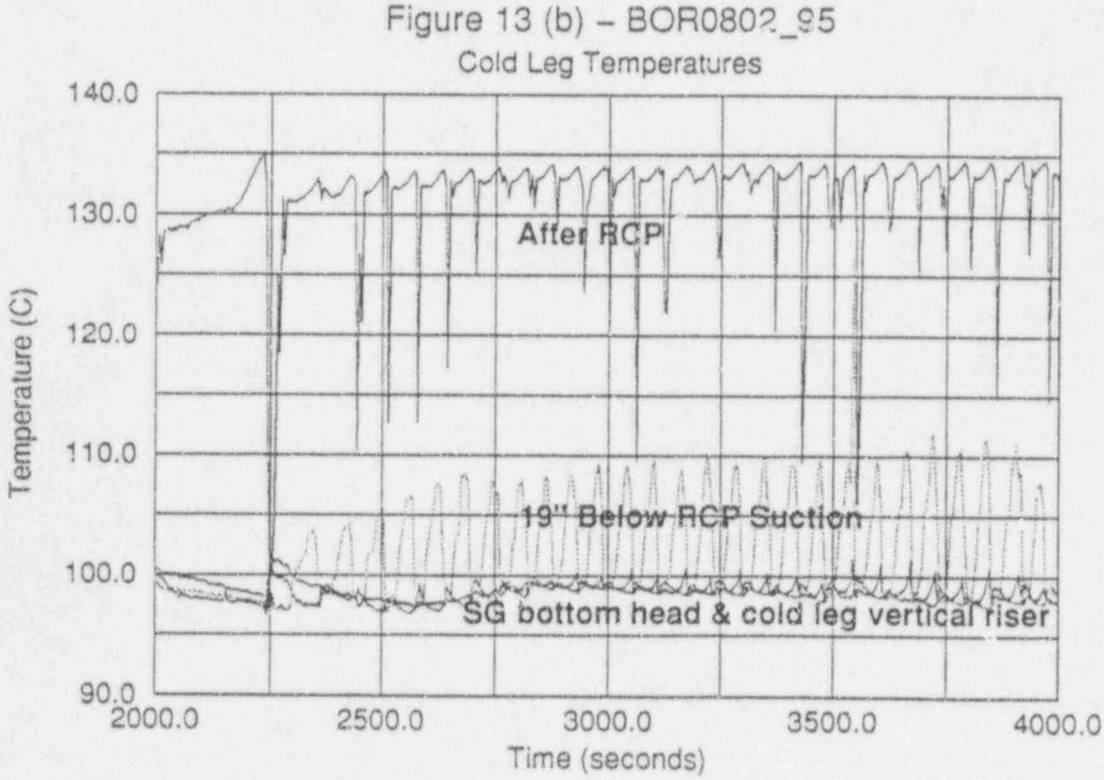


Figure 13 (b) - BOR0802_95
Cold Leg Temperatures
140.0
130.0
120.0
110.0
100.0
90.0
Temperature (C)
After RCP
19" Below RCP Suction
SG bottom head & cold leg vertical riser
2000.0
2500.0
3000.0
3500.0
4000.0
Time (seconds)

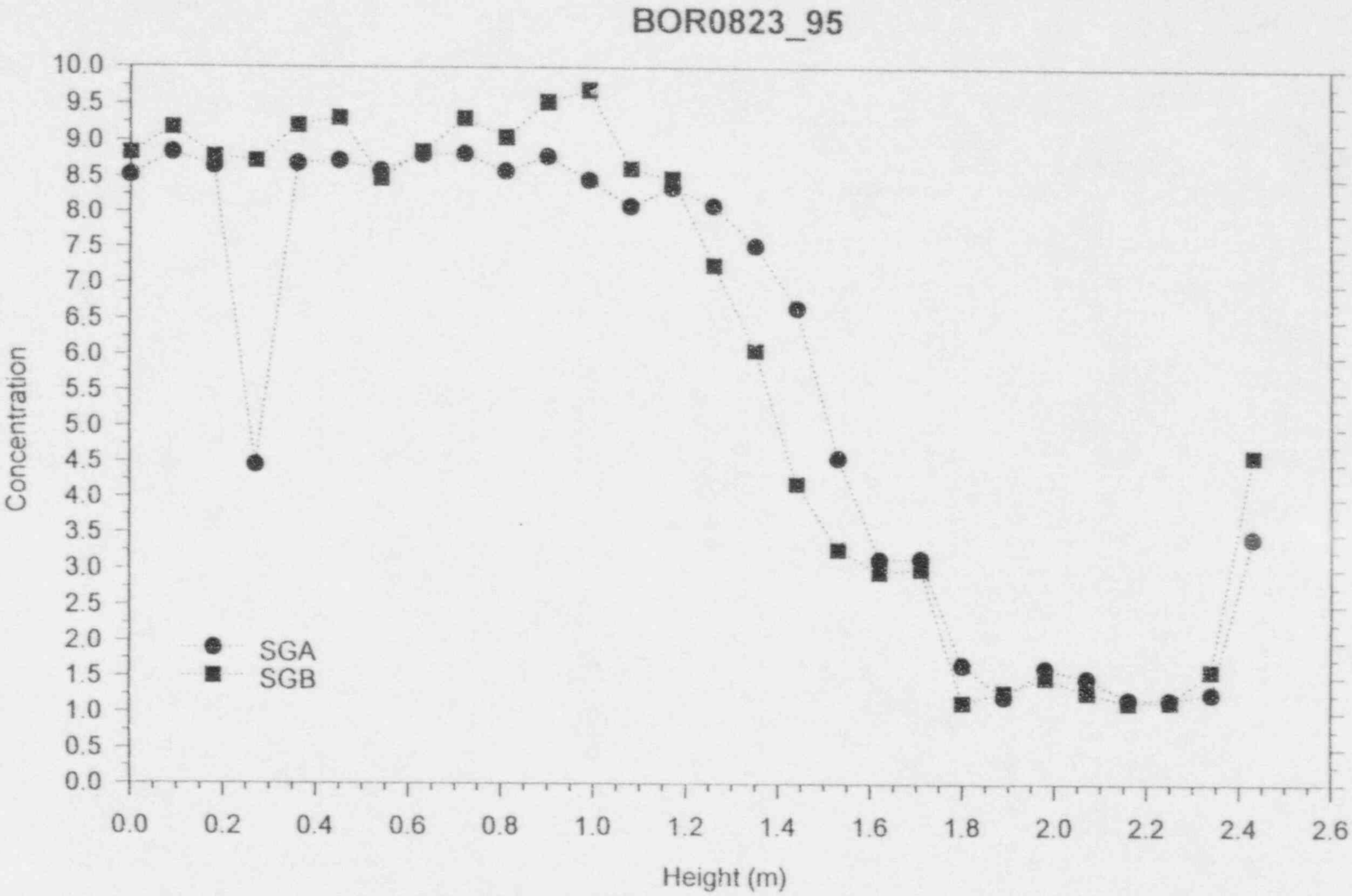


Figure 14 Experimental Salt Concentration in OTSG after Extended BCM

Figure 15 (a) - BOR0823_95

System Pressure

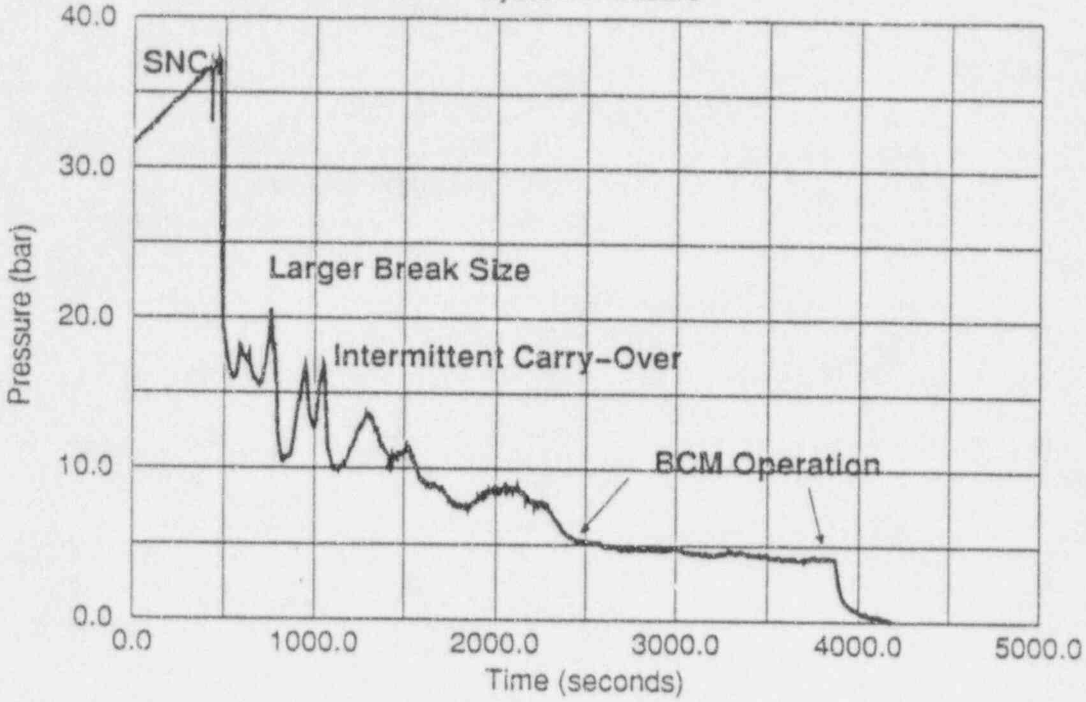
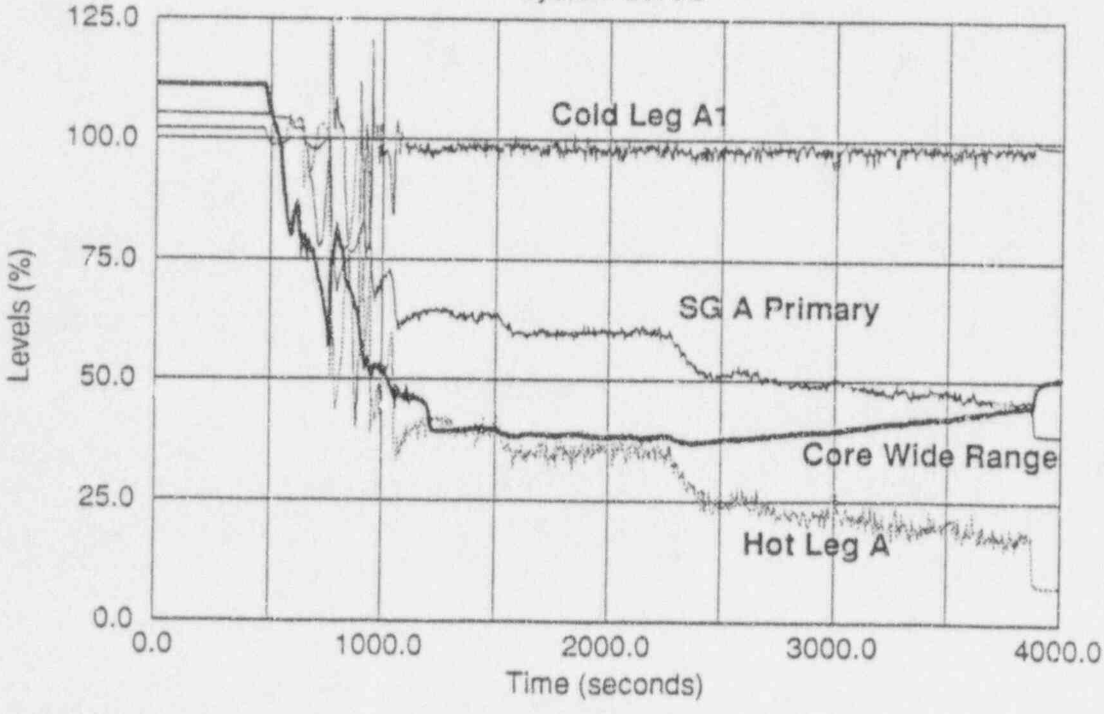


Figure 15 (b) - BOR0823_95

System Levels



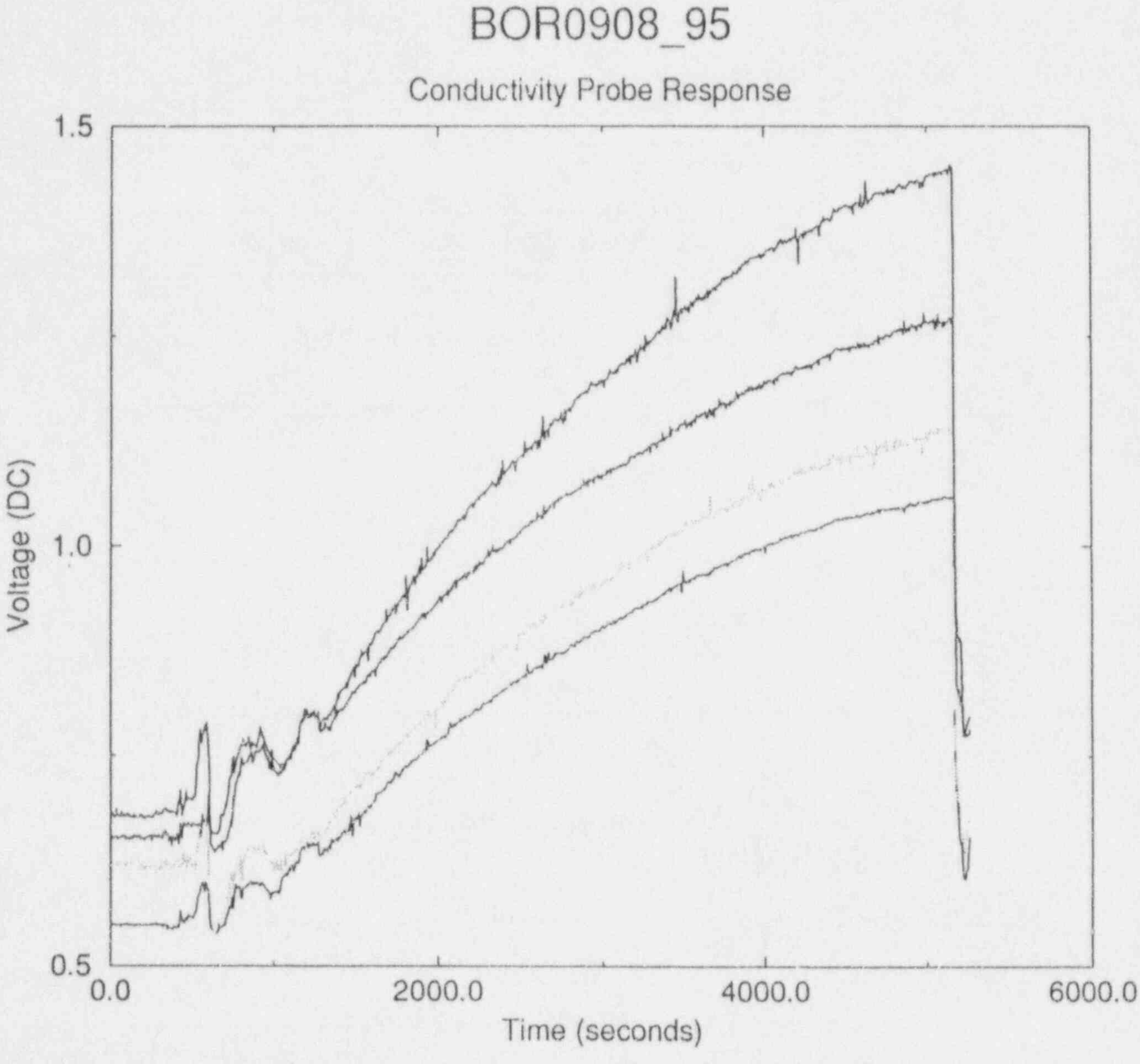


Figure 16 Lower Plenum Conductivity Probe Response

Boron mixing transient in a PWR vessel. Physical studies.

D. ALVAREZ[1], F. CASES[2], A. MARTIN[1], S. STELLETTA[2]

[1] EDF/DER/LNH

[2] EDF/DE/SEPTEN

INTRODUCTION

After the Chernobyl accident, French PWR safety studies pointed out that the potential risk of a reactivity accident may exist when the primary coolant fluid entering the reactor core is cold or insufficiently borated. Potentially risky situations are those which can lead to heterogeneous boron concentration or/and to heterogeneous temperature of the primary coolant fluid.

There is an immediate risk when the ratio of the diluted or/and cold water flow rate to the primary flow rate is high. There is a delayed risk when a plug of diluted or/and cold water is sent into the core by the modification of the primary fluid flow characteristics.

A preliminary analysis on neutronic consequences of the entering of cold and unborated water in the core has been done. By using conservative criterion, that is to say the criterion used in the rod ejection studies, and assuming a plate shape for the plug entering the core, the maximum volume allowed in the core in these conditions is estimated at 1 m^3 of cold and unborated water.

To avoid the immediate risk and the entering of more than 1 m^3 of cold and unborated water in the core, assuming that the flows between the fuel assemblies are balanced, the maximum acceptable flow rate of cold and unborated water must not be more than about 9% of the total flow rate entering the core.

A delayed risk can occur by making a plug blocking the primary flow rate in a loop or an incomplete plug, and by sending it into the core by modification of the primary fluid flow characteristics.

The volume of diluted water, before being pushed into the core, is mixed with the primary fluid in the downcomer. To prevent this volume to be more than 1 m^3 to avoid a reactivity accident, the corresponding volume in a loop must not be greater than almost 2.5 m^3.

The risky scenarios deal with the situations where the primary coolant pumps are stopped. The plug formation preferentially takes place in the lower parts of the circuit, in the vessel lower plenum or in the U pipe. A Research and Development program has been set up to study the clear and cold plug formation and its mixing with the primary coolant fluid. Three priority cases have been defined :

- Configuration P1 : Analysis of the consequences of a RCP start up when the water in the U pipe in connection with this RCP is at a heterogeneous boron concentration or temperature. The results of this study will allow us to determine the maximum of clear or cold plug volume stored in the U pipe not leading to an unacceptable situation.

- Configuration P2 : In a hot shut-down normal operation, we analyse the consequences of the CVCS cold water fluid flow mixing in the vessel, to focus on the possibility of a cold plug formation in the vessel lower plenum.

- Configuration P3 : In a cold shut-down normal operation, when the reactor cooling is performed by the Residual Heat Removal System, we analyse the possibility of a cold plug formation in the vessel lower plenum.

The results of the R&D program will allow us to :

- confirm that there are no risks of reactivity accident when a plug of 2,5 m^3 is sent into the core,

- estimate the exact acceptable volume in the loops, by studying a range of plug volumes,

- evaluate the plug formation in the vessel lower plenum.

For the moment, modifications have been made in the nuclear power plant to avoid these risky situations, on the basis of an acceptable volume plug of 2,5 m^3.

The R&D action, aiming at gaining more knowledge on vessel thermalhydraulics, consists of two complementary approaches based on mock-up experiments and numerical simulations. The overalls of this action was previously presented in the NURETH-5 Meeting [1] and the first results concerning the P1 configuration in the NUTHOS-4 Meeting [2].

Maintenance scenarios studies began in 1995. They have been performed solely with the FEM CFD code N3S. The FEM model take into account the U pipe, the primary pump and the cold leg. This mesh can be connected to the vessel mesh used in the study of the previous configurations [1] [2]. The first case in progress concerns the influence of the start-up of a boron unsaturated demineralizer. The study concerns the plug formation in the U pipe involved by the clear and cold seal injection water entering the primary circuit. At the end of the diluted water injection the primary pump is started up and the U pipe fluid is sent in the reactor vessel.

This paper presents first the CPY 900 MW PWR vessel taken into account in these physical studies, with a special focus on the geometric peculiarities. Then the 1/5th scale BORA-BORA mock-up and the 3D FEM Thermal Hydraulic code N3S are described. The results obtained until now are presented. A first set concerns a validation step performed before the studies on the three priority cases mentioned above. After that we will give the results and the degree of achievement of the studies on the three priority cases. The maintenance scenarios studies have just been scheduled in the R&D program so this paper does not present them.

The 900MW CPY PWR

As mentioned before, the purpose of these studies concerns the primary coolant mixing capabilities in French PWR's vessels. From an overall point of view we choose the 900MW CPY plant because it is a three loop type reactor which is the more common running plant in our country and then because the CPY type offers the most geometric complexities such as the downcomer's four sectorized thermal shields (Fig 1) with regards to the in vessel fluid flow mixing.

With the benefits of previous works on PWR's, especially the CEA/EDF/FRAMATOME/WESTINGHOUSE Joint Research Program PWS 2-9, we take into account as much as possible the exact vessel geometry. For example if we consider only the flow field in the downcomer, important geometric features are to be considered. Firstly the elbowed cold legs which involve an unsymmetric flow field in the downcomer inlet ; then the geometry of the thermal shields which drive the flow (the thickness of the thermal barrier is one third of the current downcomer thickness) and in correlation the position of the cold leg with regards to the thermal shields position.

The simulation area of interest starts several cold legs diameters upstream the vessel entrance elbows, up to the lower core plate (See Fig 1 for the reactor and the geometries of the study). The core itself is not simulated and the hot legs are only simulated by the space required for them in the downcomer.

The BORA-BORA mock-up

The BORA-BORA mock-up was designed in 1991 and has been running since march 1992. The three cold legs, the outer vessel itself, the lower plenum instrumentation columns and plates are made of Perspex for flow fields visualisations and for further Laser Doppler Velocimetry measurements (Fig 2).

For the hydraulic experiments, the primary flow mixing characteristics are studied at the core inlet in the mock-up close to the lower core plate by means of temperature tracking. For the steady state fluid flow mixing tests the temperature of one of the cold legs is warmed 10°C hotter than the others. For the transient discharge study of a clear plug in the vessel (Configuration P1), the plug is simulated by a warmed water plug. For configurations P2 and P3, the density effects are simulated with salted water for the cold loop and clear water for the two others. The temperature tracking technique is also used warming up the salted water.

The vessel mixing characteristics are determined between the cold legs and the core inlet plate. A temperature rake gives the reference temperature several diameters before the cold leg elbow and 60 sensors regularly spaced measure the temperature at the core inlet. In the experiments two instrumentations have been used. The first one involves 1 mm thermistor sensors and the second 0,25 thermocouples. The measured time response is 450 ms for the thermistors and 7 ms for the thermocouples. Thermocouples has been used to confirm the results obtained on the transient plug mixing in the vessel. The calibration of the sensors has been performed with an oil bath leading to a 0.1°C accuracy. By computing the total energy balance between the inlet and the outlet of the vessel gives the global accuracy of a run. A run is taken into account only when this global accuracy does not exceed 10%.

The N3S code

The FEM code N3S has been developed by the Research Branch of EDF for thermalhydraulics studies in nuclear engineering design [3] [4] taking advantage of our experience on 3D finite difference codes. The development of N3S started in 1982. The main feature is the use of unstructured meshes for complex geometries modelling. After intensive testing required by EDF Quality Assurance policy, it is now available for use as a general purpose tool which has been applied successfully to a wide variety of incompressible laminar or turbulent flows [5] with or without heat transfer [6]. For code assessment a wide range of computer program validation are made under a Quality Assurance procedure for every major release of N3S [7]. Code results are compared with analytical solutions when available or with literature experiments [8]. In an other hand our contribution to the computation of benchmark exercises proposed for international numerical workshops constitute a code validation [9].

An important work has been done on the choice of efficient algorithms and on their implementation in order to reduce CPU time and memory allocation [10] leading to release 3.1 of N3S. Table 1 gives an illustration of this work, because we begun our computations with release 3.0 and then went on with release 3.1.

N3S release	3.0	3.1	Ratio 3.0/3.1	
CPU (ms/Dt/node)	1.61	0.41	3.93	0.18
Memory (MWords) disk	355	79	4.5	171
in core storage	55	43	1.28	116
CRAY type	Y-MP			C98
Mesh (lower internals)	without			with
nb of velocity nodes	200 000			360 000

Table 1 : CPU time and Memory allocation for the steady state computation of the primary coolant flow in the vessel.

The N3S package contains four main steps : the pre-processor, the solver's interface, the solver N3S itself and the post-processor.

For the pre-processing task, the I-DEAS™ software and most exactly Object Modelling (OM) and Finite Element Modelling (FEM) have been used for the geometry definition (mapped meshing) and SIMAIL™ (free meshing) for the unstructured mesh generation.

The solver's interface PREN3S checks the mesh and prescribes the boundary conditions.

The solver N3S solves the Reynolds Averaged Navier-Stokes Equations for an unsteady incompressible or compressible flow with a standard k-ε two equations turbulence model [11]. For buoyancy driven flows with small temperature differences, the momentum equation is coupled to the energy equation using Boussinesq approximation. To take into account more important thermal effects this leads to consider Navier-Stokes equations in which density depends on temperature (varying density) : $\rho = \rho(T)$. The time discretization is based on a fractional step method [12]. At each time step the code solves successively :

- an advection step, for the non-linear convection terms of the Navier-Stokes equation and the k and ε equations, by using the characteristics method [13].
- a diffusion step on scalar variables.
- a generalized Stokes problem for the velocity and the pressure, solved either by a Uzawa and direct method for the pressure system or a Chorin-Temam algorithm (projected gradient).

Assuming a logarithmic velocity profile, we use wall functions on the boundaries to compute the friction shear stress at each time step. Additionally, a zero flux condition is given for k which leads to the definition of a second velocity scale used to prescribe ε at the wall [14].

For the post-processing task, the GRAFN3S and ENSIGHT softwares were used for FE visualisations in fluid dynamics. Obviously, the mesh and all computed variables can be plotted but also 2D isocontours or fluid trajectories with the same interpolation method that the one used in the solver N3S. ENSIGHT allows the visualisation of 3D fields.

For the application to the CPY PWR, P1-isoP2 tetrahedrons have been used [3]. The pre-processing task led to two 3D meshes : the first one (201565 nodes and 133815 elements) with the lower plenum free of obstacles and the second one (361911 nodes and 237801 elements) with the plates and instrumentation columns (Fig 3). The results presented in this paper have been performed with both meshes.

Validation step : steady state computations and experimental results

The subject of these comparisons is a cross validation of the code and the test rig on the PWR vessel complex geometry.

For the computations, we take into account the case of the primary coolant fluid flow when the reactor cooling is assured by free convection induced by the residual power. The total core flow rate is 1050 kg s^{-1} (350 kg s^{-1} in each cold legs). The pressure is 155 bars and the fluid is isothermal at 300°C. A passive tracer (an advection-diffusion equation for a passive scalar is solved with the turbulent fluid flow quantities) is used to make sure that a converged state has been reached. At the beginning of the computation the tracer value is zero, except at the three cold legs entrance plane boundaries where the prescribed value is one. The convergence state will be reached when a value of one is obtained everywhere in the domain. A second passive tracer, prescribed only on cold leg 2, gives data about the mass mixing between the cold legs in the whole computed domain and particularly at the core inlet. Predicted fields of this tracer give boron or temperature mixing (without buoyancy effects), by homothetic considerations. Two meshes have been used; one with the instrumentation in the lower plenum and one witho ut this instrumentation (see Table 2). About 60 CPU hours have been necessary for these computations.

For the experimental tests by using the temperature tracking technique, the cold leg 2 is warmed 10°C hotter the two others. In these experiments the thermistors has been used. Written in a non dimensional form, temperature gives the fluid flow mixing at the core inlet.

Table 2 gives a matrix showing the degree of achievement of the validation action.

Lower internals ➡ Tools ⬇	without	with (CPY)
N3S	Performed	Performed
BORA-BORA	Objective	Performed

Table 2 : Degree of achievement of the validation step.

Two meshes have been used; one with the instrumentation in the lower plenum and one without this instrumentation. About 60 CPU hours have been necessary for each of these computations. Figure 4 shows the mixing of the fluid injected in cold leg 2. The isolines are plotted on the developed downcomer cylinder. Figure n°6 shows the fluid flow pattern in the lower plenum for the two meshes.

Figure 5 gives the maps of iso value of the tracers at the core inlet for the computation (with and without instrumentation in the lower plenum) and the experimental results performed with the CPY geometry. A gyration effect is observed at the core inlet and in the downcomer for the computation taking into account the lower plenum internals. To understand the relative effect of the lower plenum obstacles, experiments without them are scheduled (see table 2). Nevertheless, figure 5 shows a good agreement between the numerical and the experimental results.

Configuration P1 : unborated plug transient in the vessel

The case of dilution transient studied here is related to the mixing of a clear water plug when one RCP starts up, with a zero mass flow rate in the two others cold legs. The plug when entering the vessel has a zero boron concentration. The remaining of the primary coolant fluid has an initial value of 2000 ppm boron concentration. When the plug driven back by the RCP reaches the vessel inlet, the fluid flow has well established turbulent mixing characteristics. So in our study we consider only the transient mixing of the plug, between the vessel inlet and the core inlet, at a constant mass flow rate of 1450 kg s^{-1}. This mass flow value is determined by the transit time of the plug, between the pump and the vessel, and the RCP start up flow rate data (i.e. from 0 to 4400 kg s^{-1} in 20 s).

A first set of experiments has been performed in the CPY geometry. In those tests the thermistors and many clear water reactor volumes ranging between 5 to 100 m^3 have been used in order to obtain a wide range of variation giving us some knowledge on the plug mixing. The experiments have been also performed with a steady state initial fluid flow. Only the plug motion and the mixing of this plug have a transient behaviour like in the computations.

A second set of experiments has been performed without the lower internals. Thermocouples and a 8 m^3 plug volume have been used. Seven runs for the same plug volume have been performed. The fluid flow was initially at rest. The plug was sent in the vessel by a fast fluid flow transient.

In the computations two clear water volumes have been studied and reported in this paper (3 and 8m^3). Three steps have to be reached : firstly the steady state fluid flow pattern, secondly the clear plug simulation in the cold leg and finally, the mixing and the total disappearance of the plug in the vessel. The steady state fluid flow pattern is obtained as in the previous computational case. To simulate the plug, the passive tracer method has been used. In the cold leg, the tracer number two is prescribed for a time corresponding to the simulation of a given volume. The 8m^3 plug has been computed with the two meshes and the 3m^3 plug only with the mesh taking into account the lower internals. About 55 CPU hours have been necessary for these computations.

Table 3 gives a matrix showing the degree of achievement of Configuration P1.

Influence of the geometry (steady state initial fluid flow)			Influence of the initial fluid flow condition (without lower internals)		
Internals ➡ Tools ⬇	without	with (CPY)	Fluid flow ➡ Tools ⬇	Fast transient	Steady state
N3S	(8m^3)	(3 & 8m^3)	N3S	not scheduled	(8m^3)
BORA-BORA	(8 & 12 m^3) in progress	(9 to 100m^3)	BORA-BORA	(8m^3) in progress	(8 & 12 m^3) in progress

Table 3 : Degree of achievement of Configuration P1.

EDF has a Co-operative Agreement with Vattenfall Utveckling AB (Sweden) regarding information exchange on boron dilution transients in PWR vessels. The comparison of the results obtained by the two parties is expected to give confidence on the conclusions drawn for this subject. The experiments shown in the right part of table 3 are a part of the work defined in the Agreement. Today, the Co-operative action is under progress and it is too soon to draw the final conclusions. So, in this paper the results are given in terms of the EDF's work. Comparisons between experiments and computations are given for the CPY geometry with a steady state initial fluid flow condition (left side of table 3).

To understand the phenomenology of the plug mixing, figure n°7 gives time history of the tracer for the two clear water volumes, first to reach the steady state fluid flow and then during the transient. Figure n°8 shows the mixing of the $8m^3$ plug in the downcomer.

Figure n°9 gives comparison between computations and experiments in terms of mean dimensionless concentration at the core inlet. The time scale is also dimensionless, taking into account the global transit time in the vessel (i.e. ratio of the volume of the vessel with the flow rate). This figure shows a good agreement on the general time trends between the experiments themselves and between computational results and experiments. Comparisons in terms of the local dimensionless concentration are not given in this paper. But the localisation of the minimum local concentration and its value are different. The R&D action in progress (see table 2 & 3) will give us some more knowledge to draw final conclusions on these differences. Nevertheless, we have shown that modelling the lower internals in the computations have quite no effects on the mean concentration time history at the core inlet, but the effect is important for the local concentration localisation and value.

Configuration P2 and P3 : possibility of a cold plug formation in the vessel lower plenum.

The two configurations are related to the possibility of a cold plug formation in the vessel lower plenum when the reactor is in a shut-down operation : P2 for a hot shut down and P3 for a cold one.

For the P2 configuration the reactor is pressurized at 155 bars and cooled by natural circulation. The total core fluid flow rate initial condition is 150 kg s^{-1} (50 kg s^{-1} for each loop) and 300°C. Starting the CVCS (7 kg s^{-1} and 50°C) on the cold leg number two have the consequence to make the loop number two primary fluid flow be blocked by a thermal stratification. The natural circulation is not stopped but involves only the two other loops. The stratified fluid coming from the CVCS is pouring in the vessel. In consequence, the thermalhydraulic conditions are the following : two loops at 300°C, 75 kg s^{-1} and one stratified loop at 50°C, 7 kg s^{-1}. The question of interest is the following : *is the stratified fluid pouring in the downcomer mixed by the two other loops* and if it is not case *is there a possibility of a cold plug fo mation in the vessel lower plenum.*

For the P3 configuration the flow rates involved are quite the same but the temperature difference is only 50°C instead of the 250°C for P2. So, the analysis of the primary circuit thermalhydraulic conditions have shown that for the study of a cold plug formation, the P2 configuration is the most severe. So, for the moment the study on the P3 configuration is delayed.

In reference [1] we have drawn the possibility and limitations of mock-up thermalhydraulic experiments. When simulating strong density effects the turbulence mixing can not be simulated. So the mixing between the injected water and the primary circuit is minimized in the mock-up. Nevertheless, the P2 configuration was simulated in the mock-up by means of salted water for the cold fluid and clear water for the hot one. In spite of the poor turbulent mixing obtained in the mock-up the cold fluid was well mixed and the lower plenum temperature was at the perfect mixing temperature.

Computations using the mesh without the lower in nals has been performed in this configuration. The general trend observed in the mock-up was the same than the one obtained with N3S The cold fluid is well mixed and the lower plenum was at the perfect mixing temperature.

These results are preliminary results. They have to be taken with much attention. The action on this configuration must be completed before drawing final conclusions.

Conclusions

At EDF a lot of work has been performed to study the potentially risky situations which may lead to a reactivity accident. An ambitious R&D program was built on the subject involving refined fluid flow computations pushing away the limits of the computer code capabilities and mock-up experiments taking advantage of previous work on the subject. The program was built with a special focus on validation. That involves for the code developed under Quality Assurance to be validated with integral experiments in a very complex geometry. That involves also cross validation within different mock-ups.

To draw an overall conclusion on the mixing in a PWR vessel can be that we have to model very accurately the geometry of interest both in experiments and in computations.

Let us draw conclusions on dilution events in PWR vessels. For the clear plug mixing involved by a primary pump start up (Configuration P1), the mean hydraulic transfer function of the vessel is quite well computed by the code and given by experiments. In terms of the mean concentration at the core inlet, the results given for this configuration could be used for risk evaluation with a better realistic degree of conservatism that in the preliminary studies (see the introduction). In an other hand, results could be obtained solely by computations taking into account an other loop for the plug injection in the vessel or reverse flow in the two remaining loops. For very local considerations work has to be performed again. For the study of the possibility of a plug formation in the lower part of the vessel (Configuration P2 and P3) the R&D work is in a preliminary achievement state. It is not possible to draw confidence on this subject.

References

[1] D. Alvarez O. Daubert, L. Janvier, J.P. Schneider, "Three-dimensional calculations and experimental investigation of the primary coolant flow in a 900 MW PWR vessel." Proc of the NURETH 5 meeting, pp 586-592, Salt Lake City, USA 1991.

[2] D. Alvarez, A. Martin, J.P. Schneider, "Boron mixing transients in a 900 MW PWR vessel for a reactor start up operation." NUTHOS 4, Taïwan, 5-8 April 1994.

[3] G. Pot, Bonnin, V. Moulin, B. Thomas, "Improvement of finite element algorithms implemented in CFD code N3S for industrial turbulent applications", Proc of 8th International Conference on Numerical Methods in Laminar and Turbulent Flow, Swansea, U. K. 1993.

[4] J.P. Chabard, "Modelling of turbulent complex flows on unstructured meshes using a finite element method", ECCOMAS : Second European computational fluid dynamics conference, Proc pp. 210-219, Stuttgart, 5-8 septembre 1994.

[5] J.P. Chabard, B. Métivet, G. Pot, B. Thomas, "An Efficient Finite Element Method for the Computation of 3D Turbulent Incompressible Flows", Finite Element in Fluids, vol 8 Hemisphere Publishing Corporation Ed, 1992.

[6] B. Delenne, G. Pot, "Modelling of the turbulent thermal boundary layer implemented in CFD code N3S and applications to flows in heated rooms", 8th Int. Conf. on Numerical Methods for Thermal Problems, Swansea, U. K.,12-16 July 1993.

[7] P.L. Viollet, "Modelling turbulent fluid flows in nuclear and fossil-fired power plants", Invited lecture at the conference "Directions and Issues in Hydraulics" Iowa City, May 1995.

[8] J.P. Chabard, G. Pot, A. Martin, "Industrial applications of N3S finite element code", FED-Vol 171, pp 1-9, Advances in Finite Element Analysis in Fluid Dynamics ASME 1993.

[9] L. Leal De Sousa, "Extension of a incompressible algorithm for compressible flow calculations : validation on a transsonic flow in a bump.", Workshop ETMA : Turbulence modeling for compressible flow arising in aeronautics, 1994.

[10] G. Pot, J.P. Gregoire, L. Leal De Sousa, Y. Souffez, FEMIF 8, Proc : pp 527-536. Barcelone 20-23 septembre 1993.

[11] J.P. Chabard, G. Pot, "Turbulence modelling in finite element industrial applications", FEMIF 9, Venise, octobre 1995.

[12] A. Caruso, N. Mechitoua, "Finite element computations of turbulent diffusion flame", Proc. of the 2nd Int. Conf in Advanced Computational Methods in Heat Transfer, Milan, Italy, july 1992.

[13] K. Boukir, Y. Maday, B. Metivet, "A high order characteristics method for incompressible Navier-Stockes equations", Proc. of. Int. Conf. on Spectral and High Order Methods, Montpellier, France, 1992.

[14] T. Bidot, S. Delaroff, J.-M. Vanel, G. Monville, G. Pot, "Application of the N3S finite element code to numerical simulation around a Peugeot car 405", presented at the Basel World User Days C.F.D. 1992.

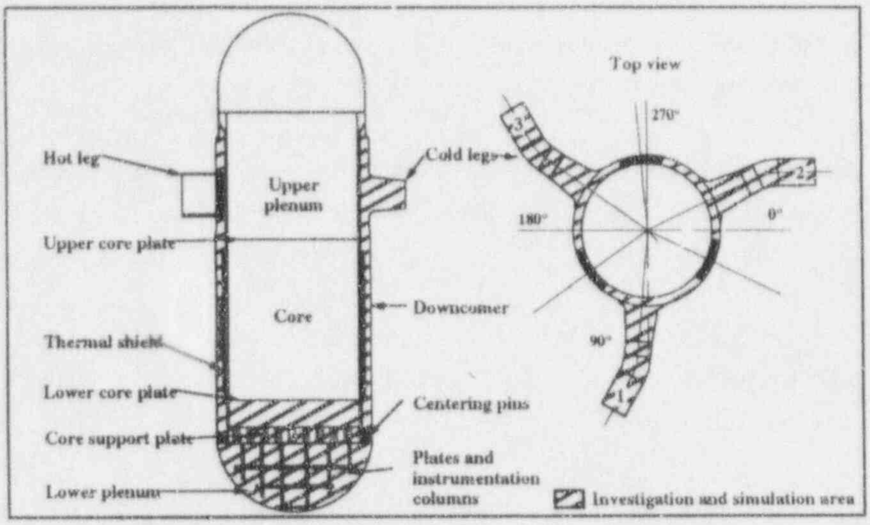


Figure 1 - 900 MW CPY PWR geometry.

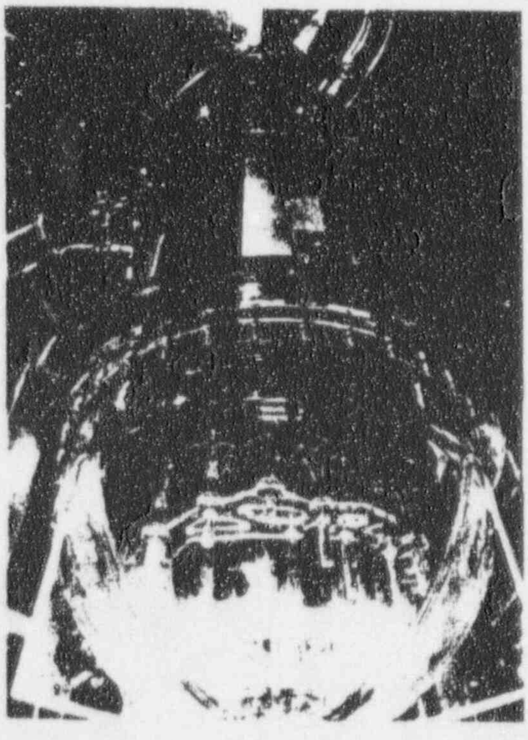


Figure 2 - BORA-BORA mock-up seen from below

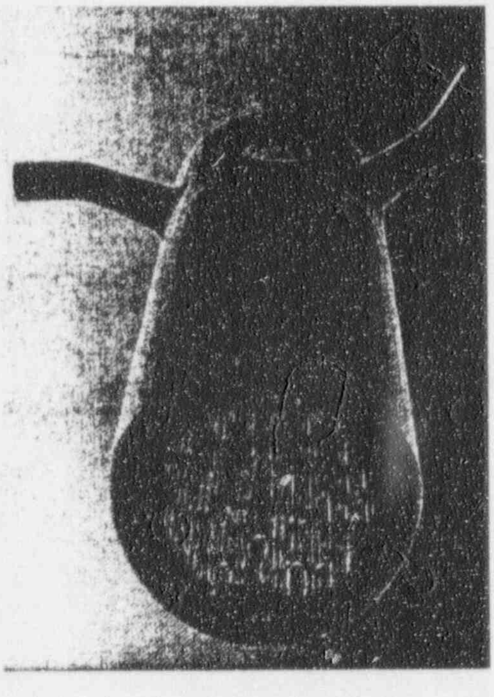


Figure 3 - Finite Element mesh seen from below

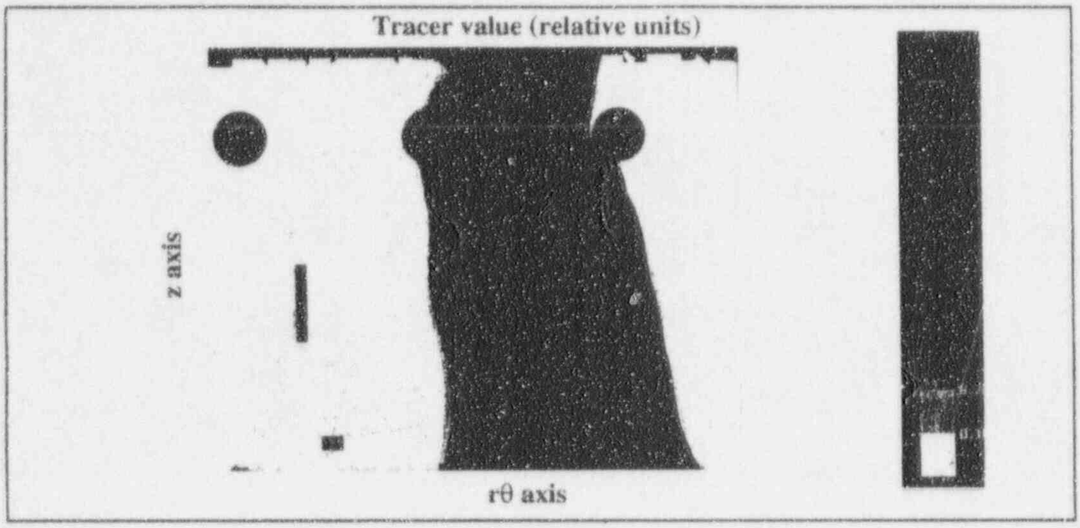


Figure 4 - Steady state mixing - N3S computed field of the passive tracer showing the mixing of the fluid coming from cold leg #2 in the downcomer.

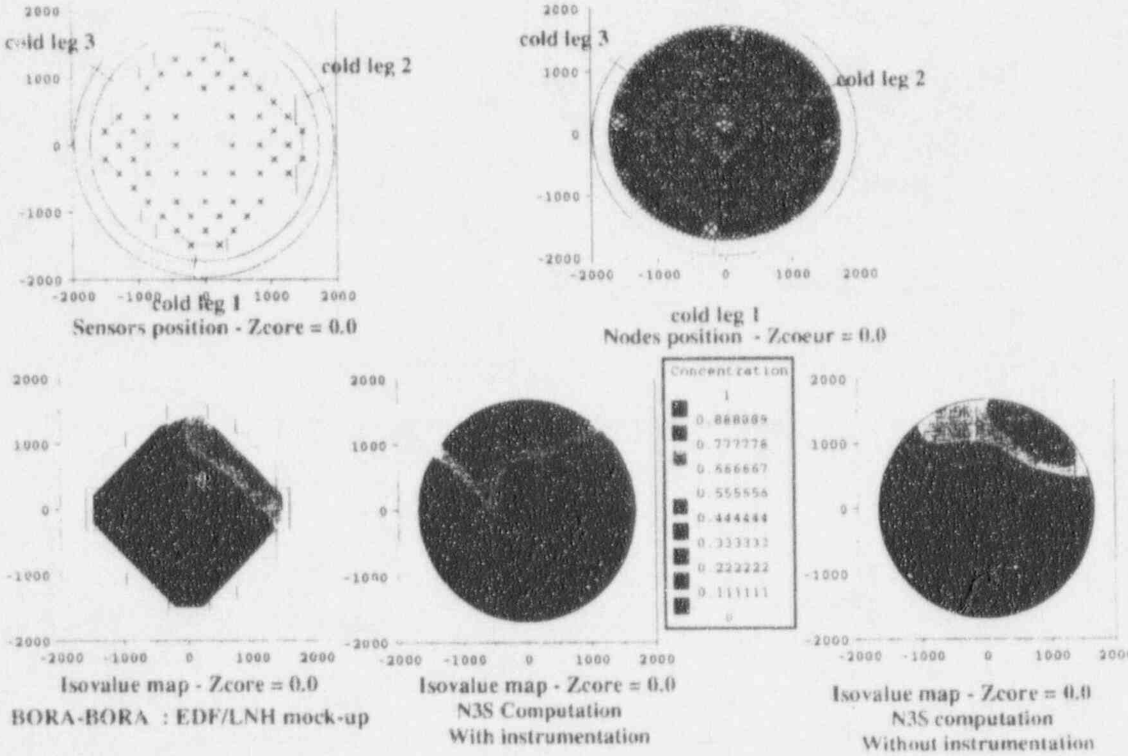
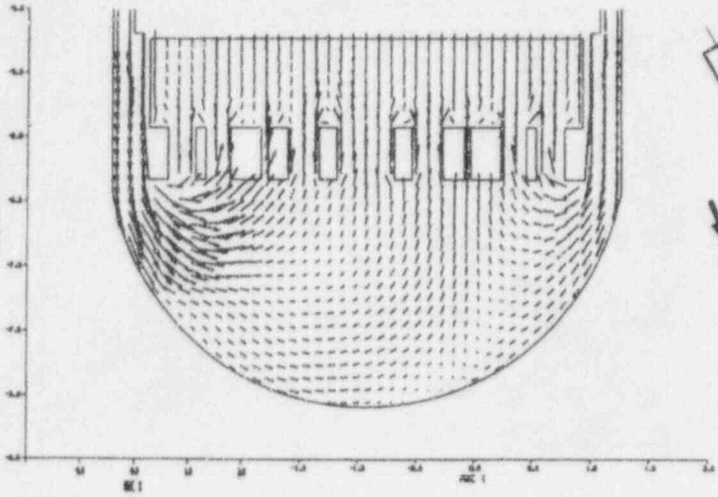
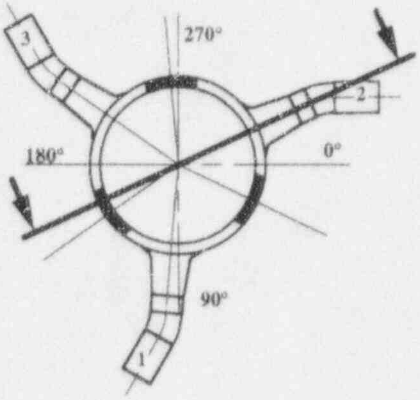


Figure 5 - Steady state mixing - Comparisons between N3S computations (two meshes) and BORA-BORA experiments on the mixing of the fluid coming from cold leg #2 in the vessel - Concentrations maps at the core inlet.





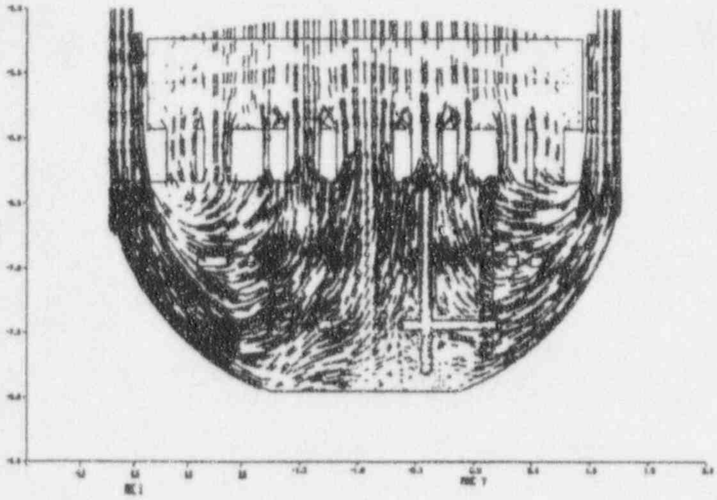


Figure 6 - Steady state mixing - N3S computed velocity plots in a vertical cut plane of the lower part of the vessel.
upper : without lower plenum inner structures
lower: with lower plenum inner structures

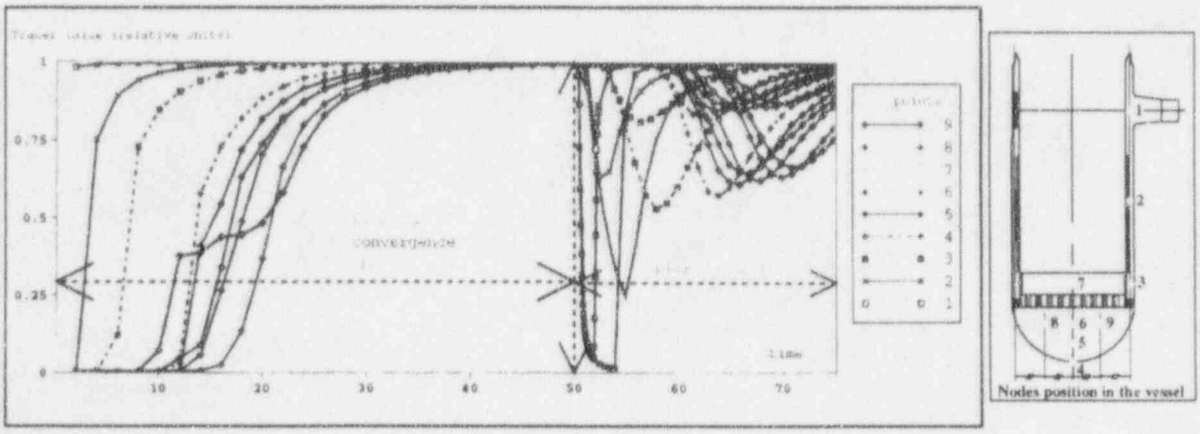


Figure 7 - Clear plug transient mixing - Tracer concentration versus time computed by N3S for several spatial locations - Illustration for two plug volumes

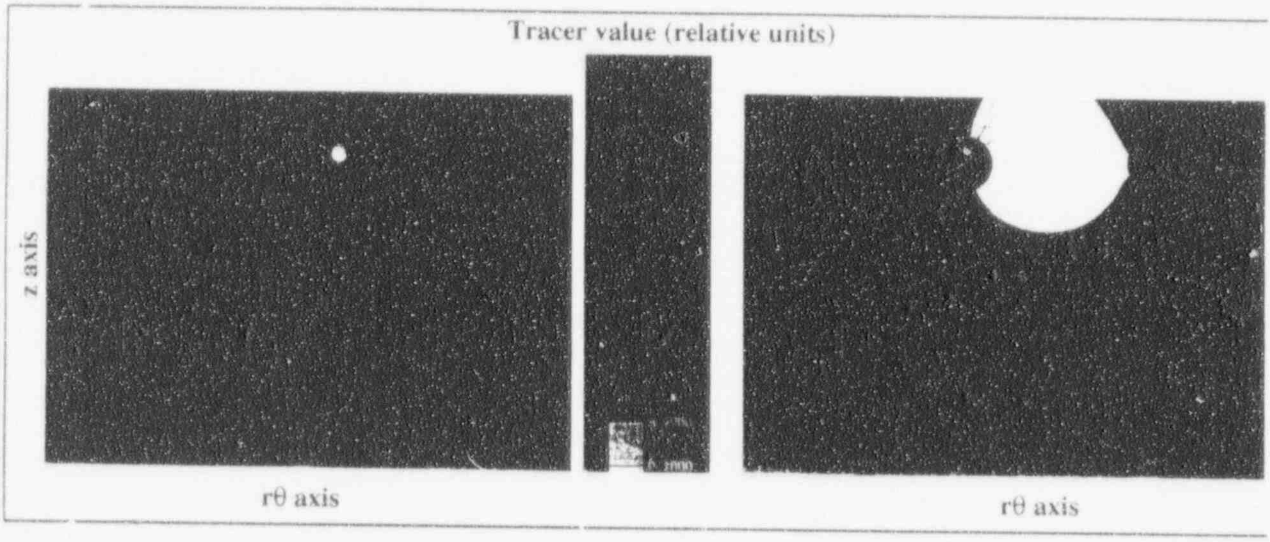


Figure 8 - Clear plug transient mixing - Concentration maps of the tracer in the downcomer computed by N3S for two different time steps.

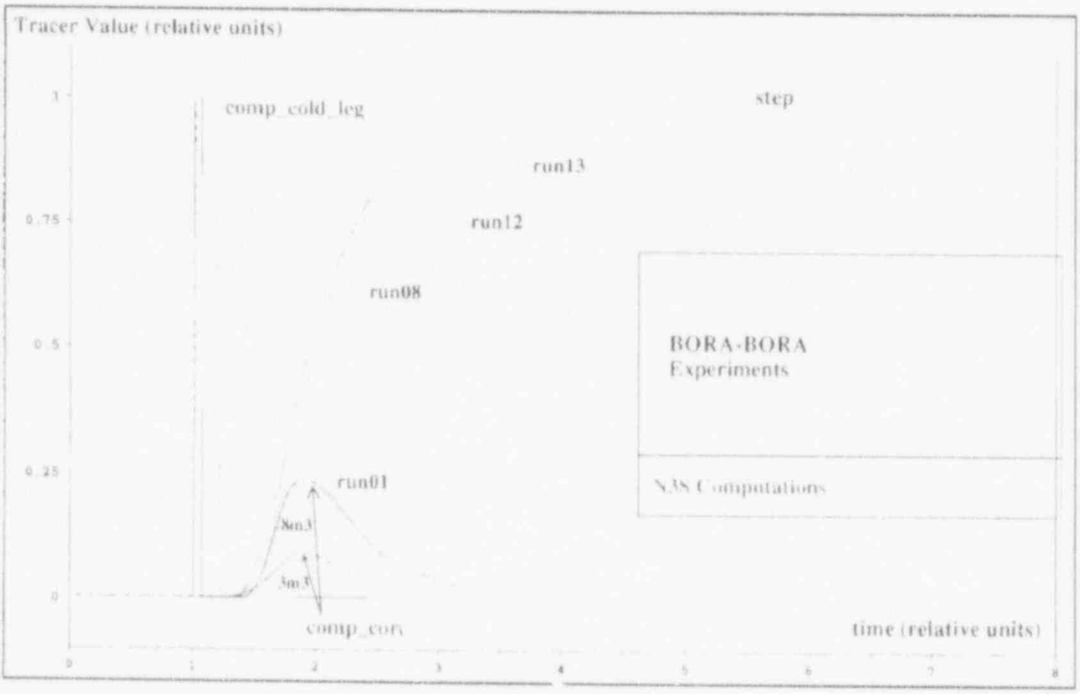


Figure 9 - Clear plug transient mixing - Mean concentration measured and computed at the core inlet.

NEUTRON RADIOGRAPHY EXPERIMENTS FOR VERIFICATION OF SOLUBLE BORON MIXING AND TRANSPORT MODELING UNDER NATURAL CIRCULATION CONDITIONS

G. Michael Morlang
Madeline Anne Feltus
Nuclear Engineering, 231 Sackett
The Pennsylvania State University
University Park, PA 16802
(814) 865-1341

Abstract

The use of neutron radiography for visualization of fluid flow through flow visualization modules has been very successful. Current experiments at the Penn State Breazeale Reactor serve to verify the mixing and transport of soluble boron under natural flow conditions as would be experienced in a pressurized water reactor.

Different flow geometries have been modeled including holes, slots, and baffles. Flow modules are constructed of aluminum box material 1 1/2 inches by 4 inches in varying lengths. An experimental flow system was built which pumps fluid to a head tank and natural circulation flow occurs from the head tank through the flow visualization module to be radiographed. The entire flow system is mounted on a portable assembly to allow placement of the flow visualization module in front of the neutron beam port. A neutron-transparent fluorinert fluid is used to simulate water at different densities. Boron is modeled by gadolinium oxide powder as a tracer element, which is placed in a mixing assembly and injected into the system a remotely operated electric valve, once the reactor is at power. The entire sequence is recorded on real-time video. Still photographs are made frame-by-frame from the video tape. Computers are used to digitally enhance the video and still photographs. The data obtained from the enhancement will be used for verification of simple geometry predictions using the TRAC and RELAP thermal-hydraulic codes.

A detailed model of a reactor vessel inlet plenum, downcomer region, flow distribution area and core inlet is being constructed to model the AP600 plenum. Successive radiography experiments of each section of the model under identical conditions will provide a complete vessel/core model for comparison with the thermal-hydraulic codes.

INTRODUCTION

Current best estimate thermal hydraulic codes for boron transport models have not been benchmarked with actual physical experiments. The objective of the current research is to develop a reactor vessel model which can be used to simulate natural circulation and low flow conditions during events which would require boron injection. The boron mixing in the model will be visualized using neutron radiography and real-time imaging systems. The Penn State Radiation Science and Engineering center has a fully equipped neutron beam laboratory[1]. The quantitative analysis of data obtained will be used to benchmark the thermal hydraulic codes.

The efforts to date have centered around construction of a system which will provide sufficient flow, allow trace element injection, and provide for recovery/reuse of the neutron transparent fluid which is extremely expensive. In addition, real-time imaging with sufficient resolution was tested. This real-time video will be digitally enhanced using computer software and the data obtained will be compared with that generated by computer code simulations of the geometries.

DESCRIPTION OF THE EXPERIMENTAL FLUID FLOW SYSTEM

The fluid flow system, as well as meeting the requirements above, had to be portable. This allowed the placement of the system in front of the neutron beam port while actually conducting experiments, and its removal for storage. A metal frame with casters was constructed to support the flow system. Figure 1 illustrates the major components of the fluid flow system.

The supply reservoir is a 5-gallon steel tank which supplies fluid to the pump, collects overflow from the head tank and collects fluid from the flow visualization module. A small centrifugal pump supplies fluid to the head tank which fills and allows gravity flow through the flow visualization module. Throttle valves are located at the outlet of the pump and flow visualization module. Level in the head tank is monitored through the use of remote controlled video cameras arranged to view the head tank gageglass. The trace element injection is controlled through the use of a remotely controlled electric injection valve. This remote control is required because of the inaccessibility of the neutron beam port area during actual neutron radiography. The system drain valve is used to recover the transparent fluid. All components are connected with 1 inch inside diameter plastic Tygon tubing. Aluminum pipe is used in locations which may experience excessive pressure. The tubing for the trace element injection line is 1/8 inch inside diameter. The head tank, supply reservoir, flow visualization module and numerous fittings were constructed at the Penn State Breazeale Reactor machine shop.

EXPERIMENTAL FLUID FLOW SYSTEM MODIFICATIONS

The most significant change to the fluid flow system has been the addition of an elevated continuous mixing system for the injection of the gadolinium oxide tracer element. Previously, the gadolinium oxide and fluorinert fluid were mixed and placed in an injection tank prior to reactor startup. The delay encountered during positioning of the reactor and reactor startup caused the gadolinium oxide tracer element to settle and clog the injection line. To eliminate this problem, an aluminum mixing chamber was built to fit the top of a standard magnetic stirring assembly. This assembly was mounted on the roof of the beam port shielding area. The mixing assembly is connected to the injection valve using thick wall 1/4 inch plastic tubing. The use of the mixing assembly allows placement of the gadolinium oxide and fluorinert fluid mix into the reservoir at any time prior to the startup of the reactor. The tubing connecting the mixing chamber and injection valve is filled with clean fluorinert fluid prior to placing the mix in the reservoir. The clean fluid eliminates any entrapped air from entering the flow visualization module as well as, keeping the mix from settling in the tubing between the mixing reservoir and injection valve.

TRACE ELEMENTS AND TRANSPARENT FLUID

The use of radiography techniques requires that the fluid in use for the bulk mainstream flow be transparent to neutrons. Trace elements injected into the mainstream flow must have a high cross section for neutron absorption.

The transparent fluids being used are the 3M Corporation's "fluorinert" family of electronic liquids, more specifically, 3M-FC-70 and 3M-FC-77. These fluorinert fluids have been found to be the most transparent to neutrons. These fluorinert fluids are extremely expensive (3M-FC-70 is $800 per gallon). For this reason, the 3M-FC-70 fluid was used initially because of its availability. Extensive work with the 3M Corporation chemist, Gregg Sherwood, has led to the determination that 3M-FC-77 would be a better choice for simulating water properties. The 3M-FC-77 fluid has properties which closely match those of water. The viscosity of 3M-FC-77 is 0.88E-6 m^2/sec compared to 1E-6 m^2/sec. One drawback is that evaporation must be considered since the fluid is exposed to the atmosphere during experimentation and recovery filtering. The fluorinert fluids with lower viscosities, such as 3M-FC-77, have an inherently lower temperature of vaporization. The 3M corporation has graciously donated 10 gallons of the 3M-FC-77 fluid.

The trace element used is gadolinium oxide powder (GdO3). The gadolinium oxide powder was mixed with a small amount of the fluorinert fluid prior to injection into the mainstream flow. The neutron opaque gadolinium oxide powder is seen as a dark streak in the mainstream flow when viewed with the real-time imaging equipment. Once an experiment is complete, the fluorinert fluid contaminated with gadolinium oxide powder is drained from the system. The extremely high cost of the fluorinert fluid requires that it be processed for reuse. A filtration system is used to remove the gadolinium oxide powder from the fluorinert fluid. Figure 2 illustrates the major components of the filtration system.

The filter assemblies are standard medical plastic filter containers with a 20 micron filter element. These containers are connected in parallel to a vacuum pump. The filter assembly is capable of handling 8 filter containers simultaneously.

The filtration rate is 1 gallon per hour. Approximately 5 hours are required for filtration following each experiment. The filter elements can be repeatedly cleaned for reuse. Occasional cracking of the filter medium was experienced after cleaning when they dried. This required the element to be replaced.

FLUID FLOW SYSTEM OPERATION

Obviously, a considerable amount of work has been done constructing and testing a suitable fluid flow system. Problems were experienced during development with leaks, material deterioration, and the entrapment of air.

The fluid flow system is initially filled with 3 to 4 gallons of fluorinert fluid. This is accomplished by removing the overflow line from the supply reservoir (Fig. 1). The overflow line is reinstalled and the supply pump started. The supply pump is controlled remotely from the area outside the neutron beam port in case rapid shutdown is required. This would be necessary in the event a fluid leak may develop. The pump outlet throttle valve is positioned to maintain a level in the head tank above the flow visualization module inlet. This provides a constant gravity flow through the module. The flow visualization module is vented at either end through vent valves. Initial tests of the module exhibited entrapped air and required the installation of vent valves. A throttle valve on the outlet of the module is used to sufficiently restrict flow through the module to prevent air backflow from the supply reservoir. Once satisfactory flow has been established through the module the gadolinium oxide injection cylinder is loaded with the mix of gadolinium oxide and fluorinert fluid. This is done just prior to positioning the fluid flow system in front of the neutron beam port to minimize settling of the gadolinium oxide in the injection cylinder. Once the fluid flow system has been positioned in front of the neutron beam port, all personnel must leave the beam port area for personnel protection [1]. At this time the Penn State Breazeale Reactor (PSBR) is positioned near the deuterium oxide tank and neutron beam port. The reactor is then brought to a power of approximately 1 MW. The real-time video system is placed in operation and as the neutron flux increases flow through the visualization module can be seen on the monitor. A videocassette recorder is activated to maintain a record of the experiment. The injection of gadolinium oxide is initiated by energizing the remote controlled injection valve. The gadolinium oxide fluorinert mix is allowed to flow into the mainstream flow entering the visualization module. Evidence of the presence of the gadolinium oxide is seen on the video monitor as black streaks in the mainstream flow. The injection of gadolinium oxide may be stopped and restarted as desired. Once the experimental run is complete the PSBR is shutdown and moved away from the neutron beam port. Personnel may then enter the neutron beam port area. The fluid flow system is drained and moved to a storage area.

EXPERIMENTS

The initial phase of the project has been to establish a satisfactory fluid flow system with sufficient video resolution to allow the collection of data. Initial flow visualization module design was a simple rectangular geometry with three holes (1/8, 1/4, and 1/2 inch) placed vertical in the mainstream flow. Figure 3a and 3b are still frames from the video taken during the first experiment.

After several successful runs with this geometry the fluid visualization module was reconstructed with a 1/2 inch slot placed completely across the mainstream flow. The flow restriction in this geometry produced turbulent flow as would be expected. Figure 4 is a still frame of the video taken with this geometry.

The actual conducting of this particular experiment was delayed by the replacement of the reactor bridge support structure. This procedure was carried out from May to July, 1994. The bridge changeout required the complete defueling of the reactor and replacement of all the core support structure. The new reactor support bridge allowed for rotation of the core as well as movement across either dimension of the reactor pool. The rotation of the reactor has helped to reduce the gamma-to-neutron ratio enhancing the image contrast.

Having successfully established a working flow visualization model, efforts have turned to conducting a quantitative analysis. The current flow visualization module was modified to allow measurement of the mainstream flowrate with the tracer element present. The exterior of the flow visualization module was etched to scale and a mix of enamel and gadolinium oxide was carefully placed in these etchings. The flow module was placed at the neutron beam port and the etchings can be clearly seen on the real-time imaging video. Figure 5 shows the scale markings as seen on the video. The round area at the second mark is a 3/8ths inch rod which has been welded into the flow module as the next simple geometry test. The scale markings were placed at one inch intervals. Also added to the video was the ability to display current time to the second.

Using the combination of flow markings and accurate time measurement, it was determined that the flow rate was 1.6 gallons per minute. This determination was verified by an actual volumetric test performed by partially disassembling the test rig piping and installation of measuring devices.

Figures 6a through 6h depict the planned construction series of flow visualization modules. Modules 6a through 6c have been constructed and satisfactorily tested. Figures 6d and 6e are under construction. Construction of a fine grid mesh matching that of an actual fuel assembly has proven to be an intricate task. Visualization of flow using these models has been very successful and construction and testing of further simple geometries will detract from the pursuit of detailed analysis of the data already obtained. Following successful testing of the grid/mesh module, construction of a complete vessel model will begin.

Problems were experienced during several of the test runs with the tracer element settling in the mix chamber prior to reaching sufficient reactor power to conduct the experiment. A stirring chamber was constructed from aluminum which could be placed over a standard magnetic mixing plate. This assembly replaced the static mixing chamber on the test stand. Following restrictions placed on the reactor operating power, videotapes of other geometries have been of insufficient contrast for producing individual still photographs. The contrast problems were originally thought to be related to the use of the new mixing assembly which was built during tests to allow reactor power to be increased to 750 kw. It was thought that the mixing assembly was keeping the gadolinium oxide powder too evenly distributed in the fluid flow. Numerous tests were conducted with varying concentrations of gadolinium oxide powder in an attempt to improve the video contrast. The gadolinium oxide powder was increased to 1.5 grams/ounce in the mixing reservoir. An experiment was conducted on 5/10/95 at this maximum concentration. Large quantities of gadolinium oxide powder not evenly distributed could be seen entering the flow visualization module through the use of a remotely controlled observation camera. At this maximum concentration of gadolinium oxide powder, contrast on the real-time video was still insufficient to distinguish any mixing taking place in the flow visualization module. Until new fuel is installed in the Breazeale reactor, further experiments with more intricate flow visualization geometries will not be fruitful.

FLOW CALCULATIONS

The fluorinert fluid FC-77 has a density of 111 lb/ft^3 and weighs 14 lb/gallon. As previously reported, volumetric flow tests determined the flow through the flow visualization modules to be 1.6 gallons/minute. Using the standard mass flow rate equation (i.e., m = rho*A*V) and the 1 inch pipe size of the flow visualization module, the velocity of the fluorinert fluid is found to be 0.01667 ft/sec.

This is considerably lower than actual forced circulation flow in a reactor created by the coolant pumps which would be 10-20 ft/sec.[4] But since we are concerned about boron mixing d .ing natural circulation conditions, this is a much more reasonable numbe .

REAL-TIME VIDEO ANALYSIS

The real-time video analysis is being performed on a MacIntosh computer using the National Institute of Health software program, NIH Image. This is the same program used for the analysis of Magnetic Resonance Imaging (MRI) and CATSCAN videos. The software has the ability to acquire, display, edit, enhance, analyze, and print images. Using a frame-by-frame video cassette recorder with a search and pause feature, individual frames from the real-time video previously obtained can be transferred into computer files. The "frame-grabber" card digitizes the images. Currently images from previous experiments are being entered into the computer and calibrated to density standards. This is accomplished by using installed standards and measurement of gray levels within the image. Once the video has been enhanced measurements can be compared between areas without the gadolinium oxide tracer and those containing the tracer element. Figures 7 and 8 are sample images taken from those currently being analyzed.

The NIH Image program also has the ability to represent individual frames as "slices" and stack these slices in a 3-dimensional data set called a "stack". This will allow the overlay of several pieces of real-time video from a particular flow visualization module. This will aid in determining if the gadolinium tracer element is following the same flow path during each injection.

FUTURE WORK

Much work has been done using real-time neutron radiography as a tool for examining two-phase flow as well as digital enhancement of images obtained using real-time neutron radiography.[5,6] While waiting for the Breazeale reactor to be refueled, digital enhancement of the videos will provide data which can be compared with simple geometries processed using the TRAC and RELAP codes. Following refueling of the reactor, attention will be turned to completion of all experiments with the simple geometries and building of a combined mockup of vessel inlet plenum, distribution plate area and core inlet.

CONCLUSIONS

Although experimental progress has been slow, progress has been made on the non-test efforts. A point has been reached where any conceivable geometry can be designed, built and examined using neutron radiography. Detailed computer analysis of real-time video data obtained has begun. Following this detailed analysis, TRAC code comparisons can begin, using computer models for the simple flow geometries and tests.

VIII. REFERENCES

[1] Cimbala, J.M., et al., "Application of Neutron Radiography for Fluid Flow Visualization," Nuclear Technology, vol 81, p.435 (June 1988).

[2] Hetsroni, G., "Use of Hydraulic Models in Nuclear Reactor Design," Nuclear Science and Engineering, vol 28, p.73 (July 1966)

[3] Radcliff, T.D., et al., "Visualization of the Lower Plenum Anomaly in the Westinghouse AP600 Reactor," Nuclear Technology, vol 106, p.100 (April 1994)

[4] THOMAS D. RADCLIFF, WILLIAM S. JOHNSON, and J. ROGER PARSONS, "Visualization of the Lower Plenum Anomaly in the Westinghouse AP600 Reactor," Nuclear Technology, 100, Vol. 106, Apr. 1994.

[5] S. FUJINE, K. YONEDA, and K. KANDA, "Digital Processing to Improve Image Quality in Real-Time Neutron Radiography," Nucl. Instrum. Meth. Phys. Res., 228, 541, (1985)

[6] G. D. HARVEL and J. S. CHANG, "Determination of Time-Dependent Void Fraction Distribution in Bubbly Two-Phase Flow By a Real-Time Neutron Radiography Technique," Nuclear Technology, 132, Vol. 109, Jan. 1995

[7] "TRAC-PF1:An Advanced Best-Estimate Computer Program for Pressurized Water Reactor Analysis," Los Alamos National Laboratory report, LA-9944-MS, US Nuclear Regulatory Commission report NUREG/CR-3567 (1984).

[8] C. ALLISON (Ed.) et al., "RELAP5/MOD3" Code Manual, NUREG/CR-5535, 1990.

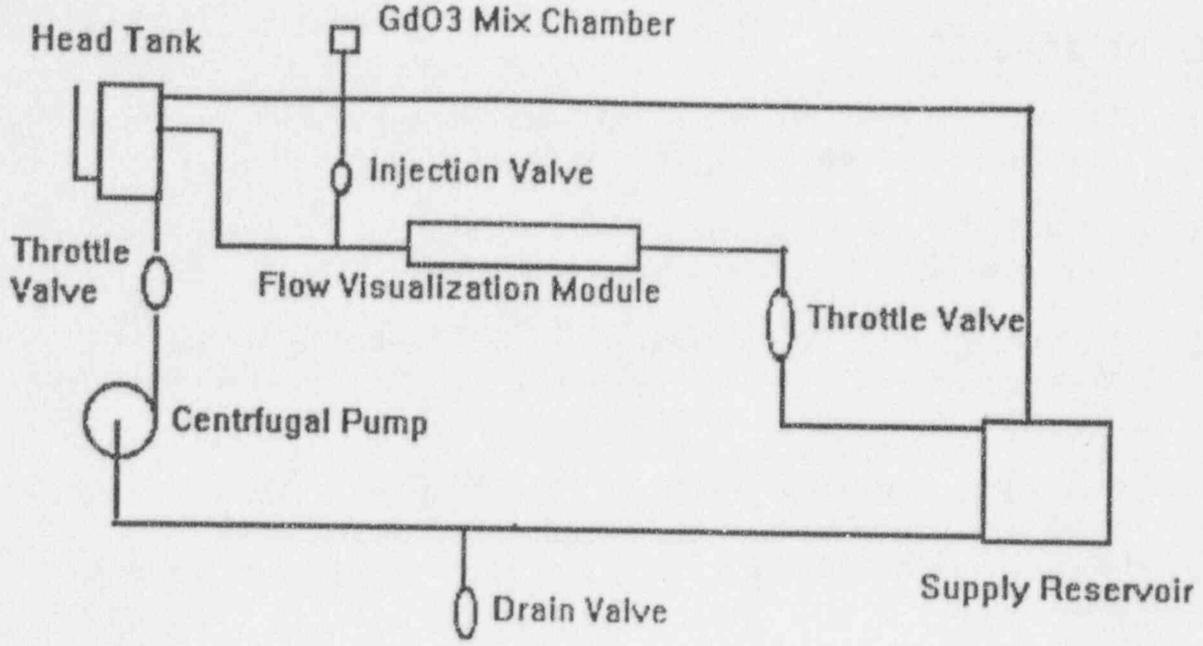


Figure 1

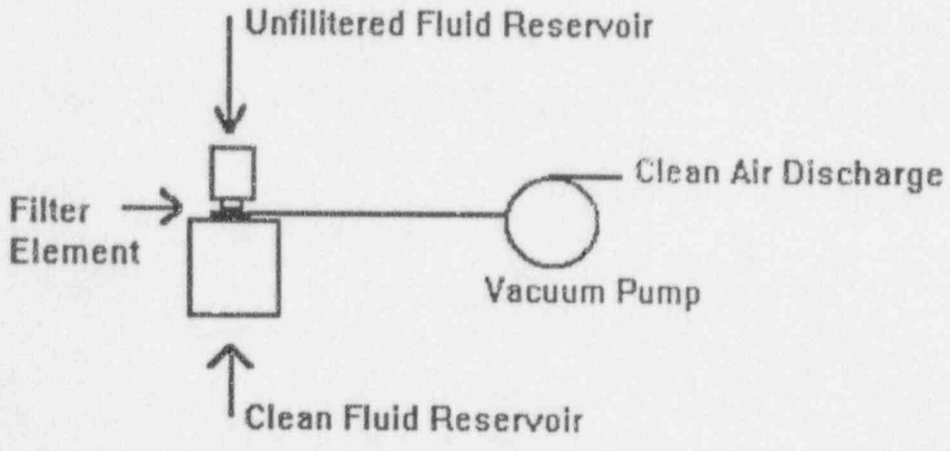


Figure 2

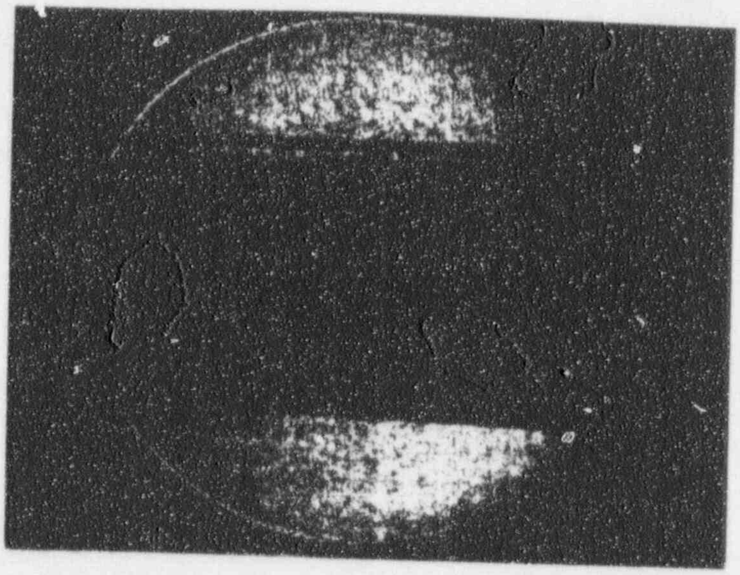


Figure 3a

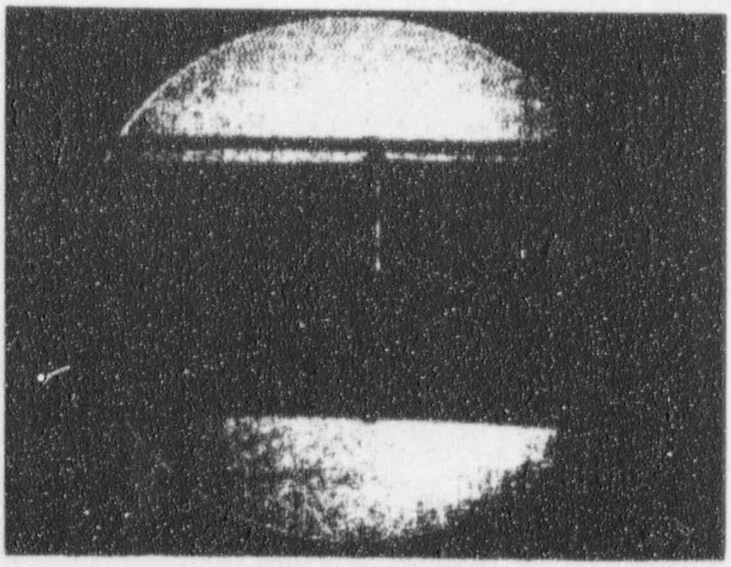


Figure 3b

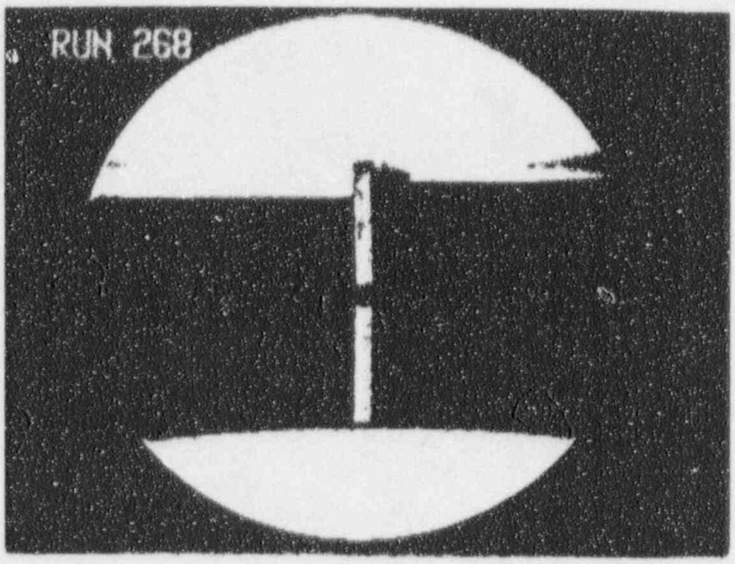


Figure 4

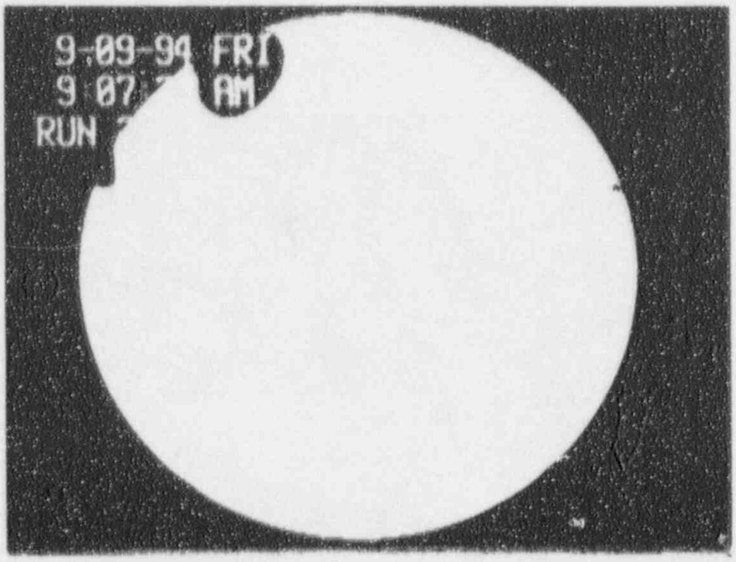


Figure 5

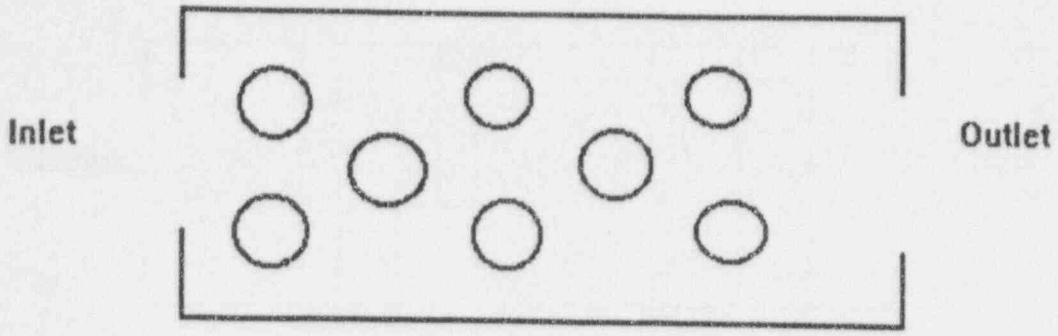


Figure 6a Pin/Lattice Structure

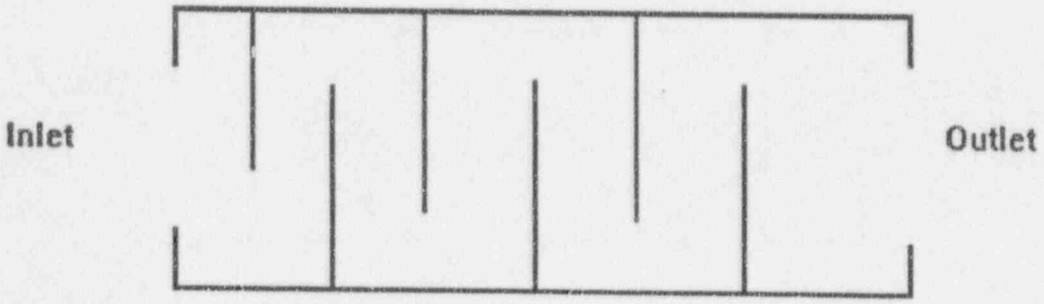


Figure 6b Baffling

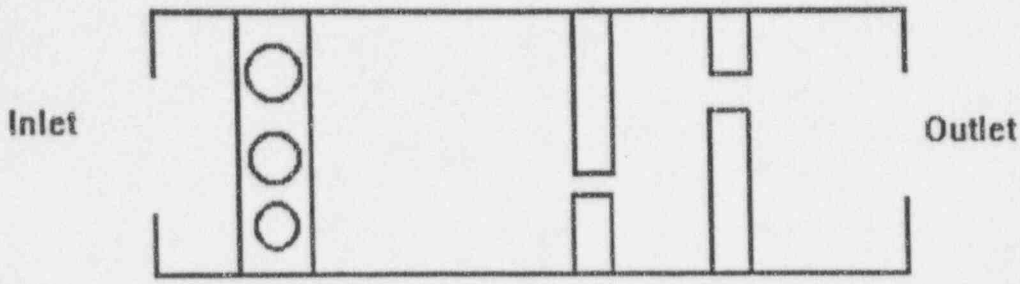


Figure 6c Slot/Gap Arrangement

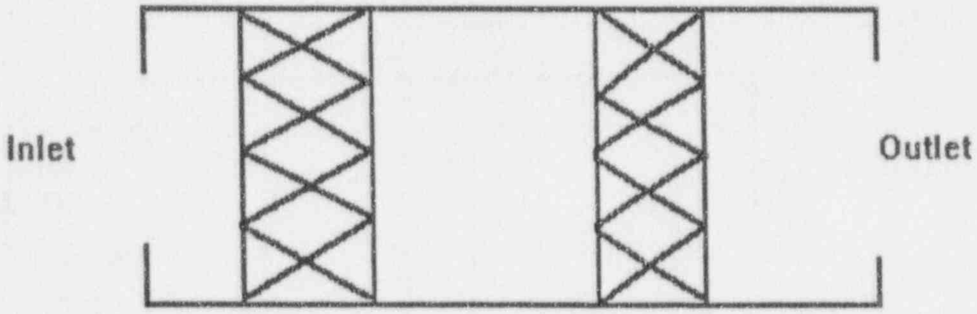


Figure 6d Mesh/Grid Strap Arrangement

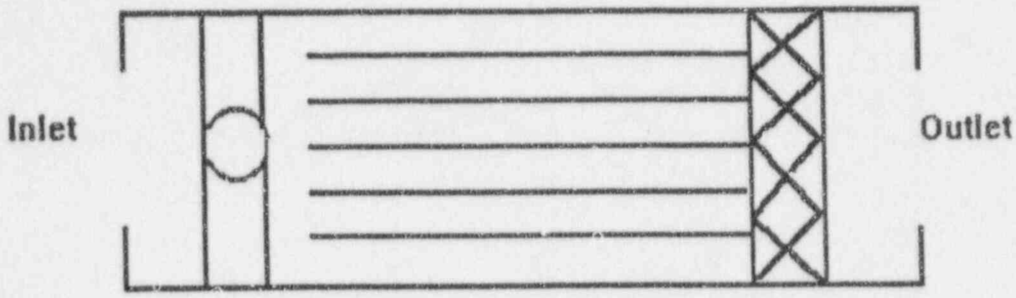


Figure 6e Fuel Pin/Grid Strap

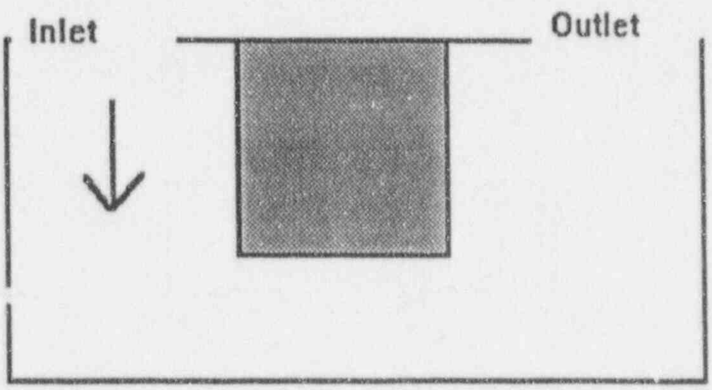


Figure 6f Sharp Downcomer Turn

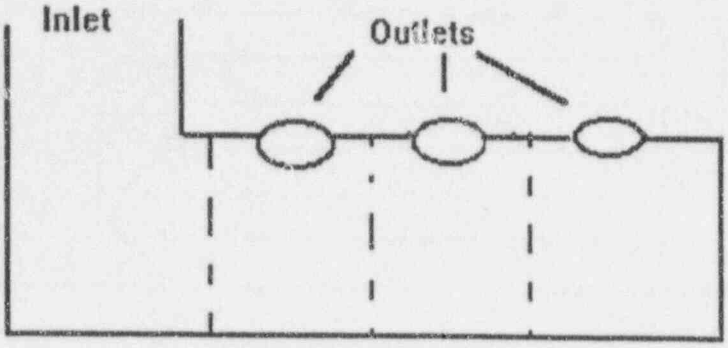


Figure 6g Inlet Plenum/Distribution Plate/Instrumentation Outlets

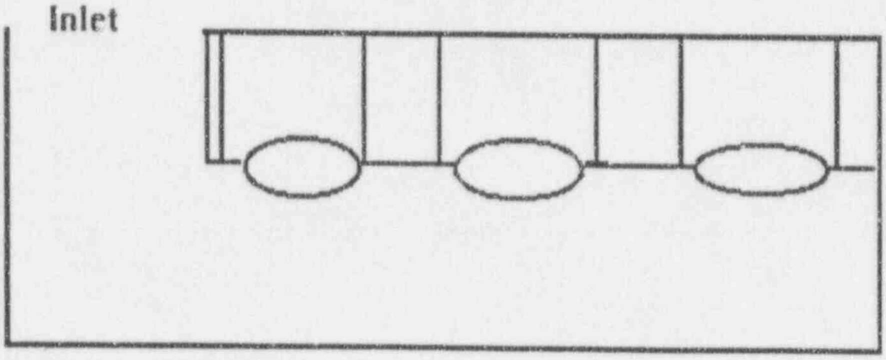


Figure 6h Downcomer/Plenum/Fuel Channels

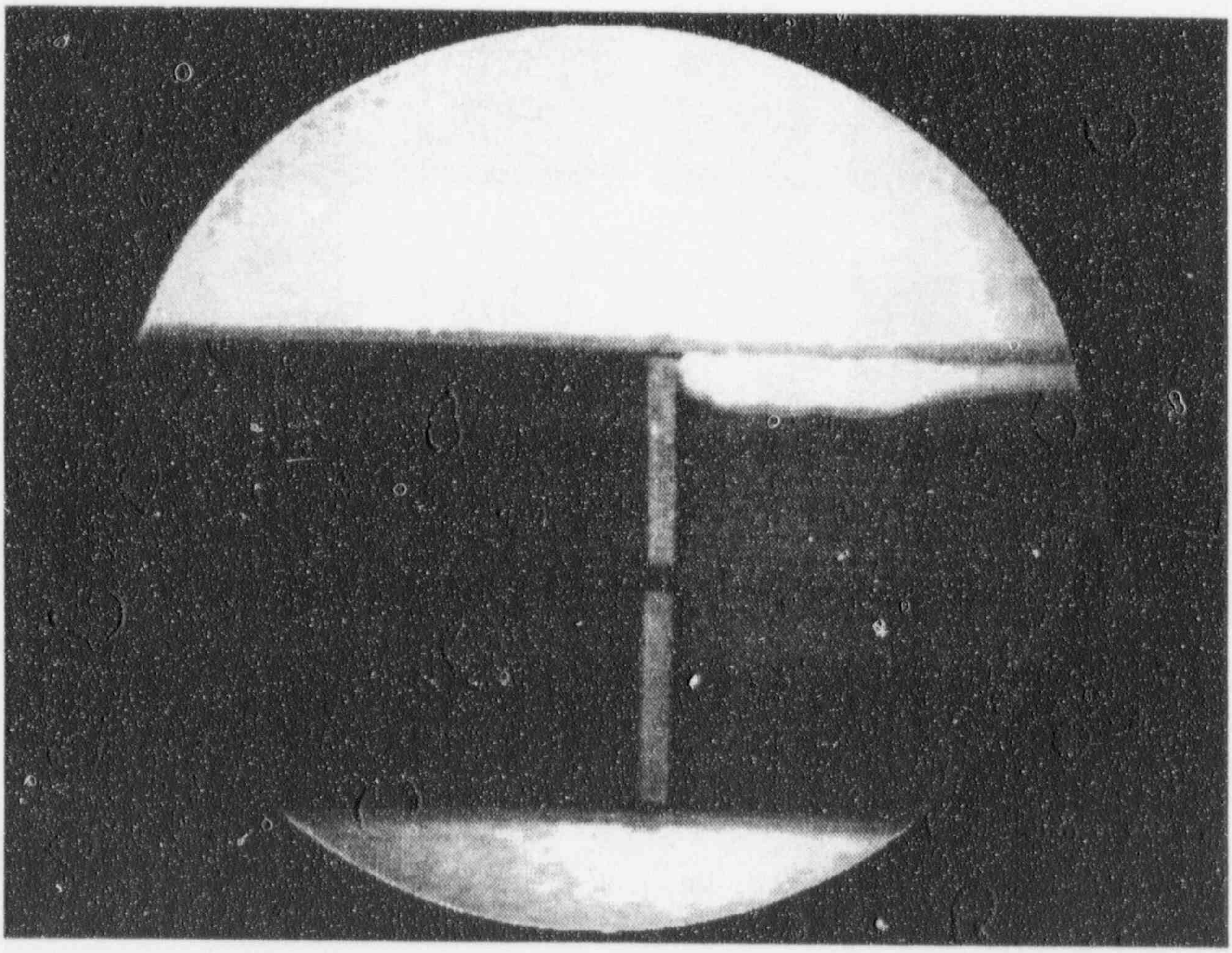


Figure 7

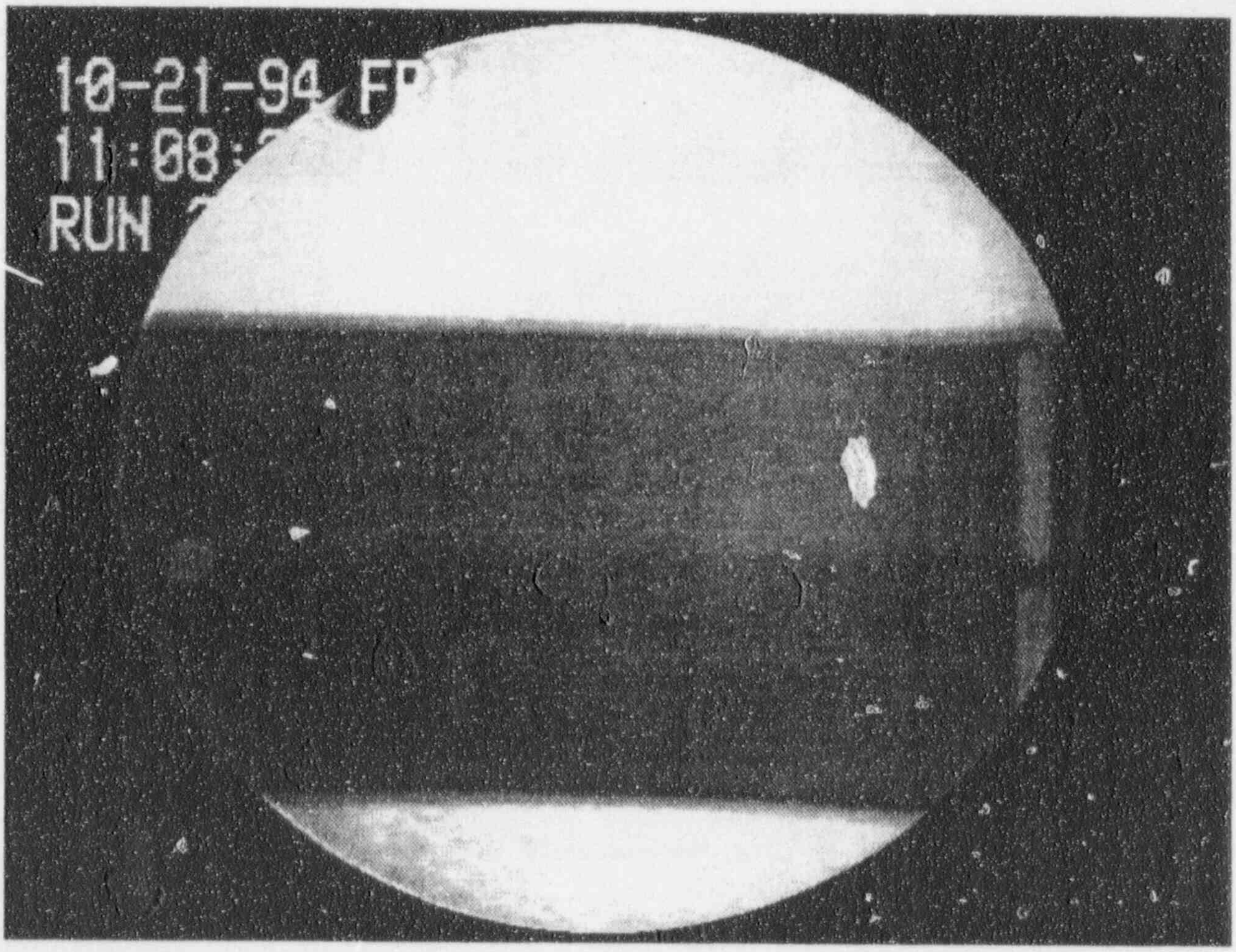


Figure 8

EXPERIMENTAL AND COMPUTATIONAL APPROACH TO INVESTIGATING RAPID BORON DILUTION TRANSIENTS IN PWRS

F. Alavyoon, B. Hemström, N.G. Andersson and R. I. Karlsson

VA. TENFALL UTVECKLING AB
S-814 26 Älvkarleby, Sweden

ABSTRACT

At Vattenfall Utveckling AB (VUAB), a 1/5 model of a three loop Westinghouse PWR has been built in order to experimentally investigate rapid boron dilution transients (RBDT). In the model, a plug of salt water solution, representing unborated water, is injected into fresh tap water circulating through the model. By measuring the conductivity of water at different points on a plane above the bottom plate, the temporal development of the salt concentration field adjacent to the core inlet has been determined. Measurements have been carried out for two different flow rates, corresponding to the steady state inlet Reynolds numbers 1.2×10^6 and 0.4×10^6. The experimental results indicate that there is a similarity between the developments of the salt concentration fields in the two cases. In other words, knowing the aforementioned field for the first case, one can deduce the concentration development of the second case on pure scaling grounds. Two conclusions may be drawn from these results: 1) For sufficiently high flow rates, the salt concentration field development is insensitive to the value of Reynolds number, and 2) The transient part of the flow rate variation, during which it is increased from zero to full flow, does not have a considerable influence on the salt field.

The research on RBDT at VUAB has been mainly experimental. Recently, however, computational work with the final aim of simulating the boron field development in full scale reactors under realistic conditions, has started. In the first step of the computational work, the flow field in a 1/5 mock-up, similar to that built for the experiments, has been investigated by using the code PHOENICS, version 1.6.6. The purpose of the computations has been to investigate the influence of computational grid on the flow field at steady state. It is shown that, for Re ~ 1.14×10^6, at least 80000 cells are required for a good resolution of only the flow field. The bottom plate and neutron pads were not modelled in these grid convergence studies. Computations on the same grid but with different Reynolds numbers (1.14×10^6 and 0.36×10^6) yield similar velocity distribution patterns.

INTRODUCTION

The present paper is an attempt to compile parts of the results of an experimental-numerical study related to rapid (local) boron dilution transients (RBDT) in PWRs which has been going on at VATTENFALL UTVECKLING AB (VUAB) during 1995.

Boron dilution transients can occur when a volume of water with very low or zero concentration of boron enters a PWR core. Severe reactivity transients may be expected if the unborated volume of water is sufficiently large. Estimating the maximum volume of unborated water which may enter the core without bringing the reactor to criticallity has attracted the attention of researchers dealing with reactor safety issues. Also other questions, such as locating the lowest concentration zones at the core inlet upon the passage of a dilute slug are of importance. An unfavourable situation is that of a dilute slug passing regions of the core where the most enriched fuel elements are. A handful of the numerical and/or experimental research done in this area are given in the list of references. For discussions of the various sequences of events that can give rise to boron dilution transients see refs [1] & [2].

The work presented here deals with the scenario in which an unborated plug of water in a reactor coolant system (RCS) pipe is pushed into the reactor vessel after starting the reactor cooling pump (RCP). The vessel configuration tested here is a 1/5 mock-up Westinghouse reactor with three loops, a main loop, through which the unborated slug is injected, and two side loops which connect the downcomer to the upper part of the core region. Two main cases can be defined based on the situation of the side loops, 1) The Ringhals case, in which the side loops are left idle and a flow through them due to pressure differences is allowed, and 2) The EDF case, in which the side loops are blocked. The present paper summarizes the main points of the research done at VUAB on the Ringhals case, refs. [5] & [13]. The EDF case has been dealt with separately and reported in ref. [4]. Similar investigations have been done by EDF, in an almost identical vessel, within the framework of a cooperative agreement with VUAB, ref. [10].

It should be remarked here that most of the research on RBDT at VUAB has been experimental. Computational activity started only recently and has been mainly focussed on grid convergence studies of the flow field at steady state, and in a geometrically simplified mock-up. The incentive for the grid convergence studies is that for the Reynolds number and time scale ranges of interest, transport of chemical species (boron in full scale and salt in mock-up) is primarily due to advection and turbulent diffusion. Therefore a correct prediction of the hydrodynamic field is the major key to a successful numerical simulation of RBDT. Further computational and experimental work is under way at VUAB and will be reported on later.

EXPERIMENTS

A 1/5 model of a Westinghouse PWR has been built at VUAB for experimental studies of RBDT. The schematic of a vertical cross-section of the model through the main inlet is shown Fig. 1. The model has a main inlet, through which water flows into the downcomer, and a main outlet, through which water flows out of the upper core region. Moreover, there are two side loops which connect the downcomer to the core, see Fig. 2. All the inlets and outlets have equal cross-sectional areas and their centers are located on the same horizontal plane. The reactor vessel model is one of the components, see Fig. 2, in the hydraulic circuit, consisting of tanks, pipes, valves, pumps etc, that is used in the experiments. Tap water with very low salt concentration, representing borated water, is stored in a 15 m^3 tank of steel. Initially, the whole system is filled with tap water. Then, the salt water solution, representing unborated water, is prepared in the salt water tank, see Fig. 2. The tap water in the pipe section between the valves V4 and V5 is drained off, and replaced by salt water. The pump P1 is started while the motor-driven valve V3 is kept closed. The experiments start when the valves V4 and V5 are opened, after which V3 starts opening. Water then flows from the tap water tank, through the valves V2, V3, V4 and V5, and the main inlet into the model. Water flows out of the model through the main outlet and the valve V7. The valve V8 is kept closed throughout the experiments, while V1 is used for regulating the flow rate.

Two series of experiments have been carried out to study the dependence of the salt concentration field on the Reynolds number. The flow rate history of the two cases is shown in Fig. 3a. The rate at which the flow is increased is the same in both cases. The flow through the side loops is caused by the pressure difference between downcomer and core, and is approximately 7%-10% of the main flow rate, see Fig. 3b. The water temperature in the experiments is 25 °C for the high and 15 °C for the low flow rate case. Based on the diameter of the main inlet pipe, and the maximum flow rate, the Reynolds numbers for the high and the low flow rates become 1.2×10^6 and 0.4×10^6, respectively. In the present work, the former case is referred to as the HRC (High Reynolds number Case) and the latter as the LRC.

The salt water plug in the experiments has a volume of 64 liters, which corresponds to 8 m^3 of unborated water in full scale. The front of the slug is initially at a distance of approximately 1.9 m from the entrance to the downcomer. The passage of the salt water plug into the downcomer is registered, see Fig. 4a, by means of probes which measure the conductivity of water at the measuring station denoted by x.2 in Fig. 2. The passage of the salt water into the core is registered by 61 probes positioned across a horizontal plane midway between the bottom plate and the core inlet, see Fig. 1. Each probe measures the conductivity of water as a function of time. One can hence get a picture of the temporal evolution of water conductivity distribution at the inlet of the core. For a given temperature and sufficiently low salt concentration, the conductivity of water can, with a high degree of accuracy, be expressed as a linear function of salt concentration, refs. [4] & [6], ie

$$\gamma=(9.786\times10^{-2}+3.602\times10^{-3}T)s \qquad (1)$$

where $\gamma(\frac{1}{\Omega m})$, s ($^0/_{00}$ or g salt per kg water) and T(°C) denote conductivity, salt concentration and temperature, respectively.

In the experiments reported here, salt is used instead of boron. Since convection and turbulent diffusion are the dominating mechanisms of species transport in the present work, it can be stated that the boron concentration field can be deduced by a proper scaling of the measured salt concentration field. In view of the linearity of expression (1), the nondimensional boron concentration can be defined as

$$c=\frac{\gamma-\gamma_u}{\gamma_b-\gamma_u}, \gamma_b \le \gamma \le \gamma_u \text{ and } 0 \le c \le 1 \qquad (2)$$

where γ_u=0.56 to 0.57 $\frac{1}{\Omega m}$ and γ_b=0.03 to 0.04 $\frac{1}{\Omega m}$ respectively denote the initial conductivities of the salt water plug (unborated) and the tap water (borated). The experimental results of this work are presented as profiles and contours av the nondimensionsal boron concentration c.

COMPUTATIONS

Most of the computational efforts at the present stage have been focussed on grid convergence studies. The commercial CFD code PHOENICS version 1.6.6 has been used for the computations of the present work. PHOENICS solves the equations for conservation of momentum, mass, heat etc using the finite volume approach. The following parts of the reactor vessel have not been modelled in this work:

1) The neutron pads in the downcomer,
2) The bottom plate,
3) The structures in the lower plenum, and
4) The fuel bundles in the core

Besides the above geometrical simplifications, it is assumed that:

1) The working fluid is incompressible and isothermal water with the density $\rho = 997.1 \frac{kg}{m^3}$ and the kinematic viscosity $\nu = 0.894 \times 10^{-6} \frac{m^2}{s}$ for the high Reynolds number case, and $\rho = 999.1 \frac{kg}{m^3}$ and $\nu = 1.138 \times 10^{-6} \frac{m^2}{s}$ for the low Reynolds number case,
2) The flow is steady and fully turbulent, and
3) The flow rate is 0.144 $\frac{m^3}{s}$ for the high Reynolds number case, and 0.0576 $\frac{m^3}{s}$ for the low Reynolds number case, through the main inlet and outlet, and 10% of that through each of the other two loops.

The turbulent flow in the present work has been modelled by the *k*-ε model. A cylindrical co-ordinate system (r, ϕ, z), with its origin at the center of the top part of the vessel, is used. The *z*-axis is vertical and directed downward. The hemispherical shape of the lower plenum is modelled by blocking appropriate number of cells on the peripheri of the mesh. Several computational cases with 40000 to 320000 cells and a Reynolds number of 1.14×10^6 have been studied. The LRC ($Re \approx 0.36 \times 10^6$ in the computations) has been studied on a coarse grid with 40000 cells. Depending on grid fineness and Re, each study case took about 5 to 15 CPU days on a 384/340 Mbytes/Mflops Silicon Graphics Challenger work station.

RESULTS and DISCUSSIONS

Experiments

The experimental results of the present work are given in Figs. 3-6 as profiles and/or contours of conductivity or nondimensional concentration. Two series of experiments have been carried out: The low Reynolds number case (LRC) and the high Reynolds number case (HRC). For each case five identical tests are made in order to make sure that the results are reproducible, and the ensemble averages more accurate. The measurements shown in the figures are, for each case, the ensemble averages of the five tests of that case.

Due to the negligible effects of buoyancy, the only two nondimensional groups that influence the measurements are the Reynolds number and the Strouhal number (defined as $\frac{\text{average velocity} \times \text{time scale}}{\text{length scale}}$), ref. [3]. As it can be seen in Fig. 3a, the flow rate for the HRC increases to 2.5 times the flow rate for the LRC in a 2.5 times shorter period of time. This is done in order to ensure equality of the Strouhal numbers for the two cases. The Reynolds number for the HRC, on the other hand, is more than three times larger than that for the LRC (due to different water temperatures). Consequently, the similarities and the discrepancies that the nondimensional concentration fields of the HRC and LRC show for corresponding values of time ($time_{LRC} = 2.5 \times time_{HRC}$), reveal the sensitivity of the results to the Reynolds number.

Fig. 4a shows the conductivity of water measured at the point x.2, see Fig. 2, which is in the main inlet at a distance of 35 cm from the downcomer. Fig. 4a shows that the initially sharp interfaces between salt water and tap water at the front and rear ends of the slug have, on their way to the downcomer, become partially diluted. This is primarily due to the action of turbulent diffusion during the time it takes for the slug to move through the pipe system to reach the downcomer. The core of the salt water plug is however still concentrated for both the HRC and the LRC. The rear end of the salt water enters the downcomer after 16 seconds for the LRC, and 6.5 (~16/2.5) seconds for the HRC. Fig. 4b shows the spatial (ensemble) average of the nondimensional concentration across the measuring section at the core inlet. The figure shows that, for the LRC (HRC), diluted salt water solution reaches the core inlet after about 20 s (8 s), decreases rapidly from 1 to a minimum value of 0.79 (0.77) and then increases, though at a somewhat slower rate, to its initial value of 1. Fig. 4 shows that, on an average basis, the LRC and HRC show similar behaviours. The question is, however, whether this similarity applies to the details of the temporal development of the concentration field. This question can be answered by comparing the distribution of the ensemble average of the concentration field at the core inlet for different values of time, see Fig. 5.

In Fig. 5, the plots on the left hand side are the HRC and the ones on the right hand side are the LRC. The plots in each row belong to corresponding values of time, and are therefore directly comparable. The figures show that the central region of the core inlet parallel with the inlet diameter responds slowly to the passage of the diluted slug, and retains a high nondimensional concentration throughout the experiments. One possible explanation of this phenomenon can be the presence of one or two separating flow regions parallel with the inlet diameter. The minimum nondimensional concentration passing the core inlet is shown in Fig. 6. The regions with lowest nondimensional concentration are observed at about half a radius from the center of the measuring section and almost parallel with the diameter passing through the outlet, see Figs. 5 & 6. The minimum nondimensional concentration is approximately 0.43, and appears after 21.7 seconds for the LRC and 8.4 seconds for the HRC.

As it can be seen in Figs. 5 and 6, there is a striking similarity between the concentration distributions of the two cases. This indicates that even the Reynolds number (Re) of the LRC is high enough to render Re-independent measurements. Consequently, one can (cautiously) conclude that the experimental results of the dilution scenario studied here in a 1/5 mock-up can, after appropriate scaling, be extended to apply to a PWR in full scale.

Computations

The computations have been focussed on grid convergence studies of the hydrodynamic field for Re $\sim 1.14\times10^6$. The case of Re $\sim 0.36\times10^6$ has also been computed on a coarse grid (40000 cells). Some results are shown in Figs. 7 & 8 as velocity vectors and filled contours of the vertical velocity component. Fig. 7 gives an unfolded picture of a vertical (cylindrical) plane in the middle of the downcomer. The radial component of velocity (perpendicular to the plane of the paper) is turned off in Fig. 7. The regions of the downcomer which intersect the main inlet are seen as sources of mass in the upper right hand sides of Figs 7a-c.

According to Figs. 7 & 8, the similarities between the flow fields shown in 7a, 7b and 7c clearly outweigh their discrepancies. The vector fields show that, upon entering the downcomer, water moves mainly in a X-shaped pattern, ie two horizontal streams to the left and to the right of the inlet, and two obliquely downward below the inlet. There is a region with a weak upward flow in the downcomer below the inlet. This is in agreement with the flow field reported by Fry [11], but in disagreement with the visualizations that were made in the mock-up in connection with the work presented in ref. [3]. Generally one can observe that the X-shaped main flow induces a number of secondary flows (vortices), rendering a complex flow pattern in the downcomer.

Computations on various grids show that a good prediction of only the flow field requires more than 80000 cells. It should be expected that computing the concentration field will require more than twice that many cells to reduce the influence of numerical viscosity. Alternatively, one can use particle tracking methods which lack diffusivity. Such methods can be considered as very conservative. One can however reduce the degree of conservatism of particle tracking by accounting for turbulent fluctuations in determining the particle tracks.

CONCLUSIONS

The case considered in this work is that of a Westinghouse PWR with three loops, two of which are left idle (Ringhals case) while starting the RCP sweeps an initially dilute slug through the RCS pipe into the reactor vessel.

The main conclusion of this study is that, for sufficiently high Reynolds numbers, and the same Strouhal number, the development of the concentration field at the core inlet in a 1/5 mock-up is Re-independent and displays similar patterns. In other words, after appropriately scaling the relevant quantities of interest, the experimental results of the dilution scenario studied here in a 1/5 mock-up can be extended to apply to PWRs in full scale.

The computational efforts of this work have been focussed on grid convergence studies. The results show that for a good resolution of the flow field in the mock-up, at least 80000 cells are required.

Acknowledgements

The present work has been financed by the Swedish Nuclear Power Inspectorate, SKI, and VATTENFALL AB, RINGHALS.

REFERENCES

[1] Jacobson, S., "Risk evaluation c. local dilution transients in a pressurized water reactor", Ph. D. thesis no. 275, Linköping Institute of Technology, Sweden, 1992.

[2] Hyvärinen, J., The inherent boron dilution mechanism in pressurized water reactors, Nuclear Engineering and Design, vol. 145, pp 227-240, 1993.

[3] Tinoco, H., Hemström, B. and Andersson, N.G., Physical modelling of a rapid boron dilution transient, report No. VU-S 93:B21, Vattenfall Utveckling AB, 1993.

[4] Hemström, B. and Andersson, N.G., Physical modelling of a rapid boron dilution transient. The EDF case., report no. VU-S 94:B16, Vattenfall Utveckling AB, 1994.

[5] Hemström, B. and Andersson, N.G., Physical modelling of a rapid boron dilution transient- I. Reynolds number sensitivity study for the Ringhals case, report no. US 95:5, Vattenfall Utveckling AB, 1995.

[6] Jacobson, S. and Sandervåg, O., A comparison between a mixing experiment in a 1/5 scale reactor vessel mock-up and a numerical simulation with the

PHOENICS code, Presented at european Nuclear Society Meeting in Avignon (France), April 1995.

[7] Andersson, N.G., Hemström, B., Karlsson, R. and Jacobson, S., Physical modelling of a rapid boron dilution transient, Proceedings of Nureth 7, Saratoga Springs, USA, 1995.

[8] Alvarez, D., Daubert, O., Janvier, L. and Schneider, J.P., Three dimensional calculations and experimental investigations of the primary coolant flow in a 900 MW PWR vessel, Proceedings of NURETH 5, Salt Lake City, USA, 1992.

[9] Alvarez, D., Martin, A. and Schneider, J.P., Boron mixing transients in a 900 MW PWR Vessel, The 4th Int. Topical Meeting on Nucl. Thermal Hydraulics, Operations and Safety, Taiwan, April 1994.

[10] Alvarez, D. Schneider, J.P., Experimental studies of the fluid mixing in PWRs; Boron dilution transients of a 8 M3 free boron plug in the vessel without the lower internals. Vattenfall Methodology, report HE-41/95/025/8, EDF, 1995.

[11] Fry, C.J., Validation of the CFD code CFDS-FLOW3D for modelling boron dilution transients during pump start-up in pressurized water reactors, AEA/CS/16401557/R1, Jan. 1995.

[12] P. Gango, Numerical boron mixing studies for Loviisa nuclear plant, Proceedings of Nureth 7, Saratoga Springs, USA, 1995.

[13] Alavyoon, F., Numerical investigations of rapid boron dilution transients for a 1/ PWR mock-up - I. Grid convergence studies of the flow field, report no. US 95:12, Vattenfall Utveckling AB, 1995.

RINGHALS PWR reactor mock-up

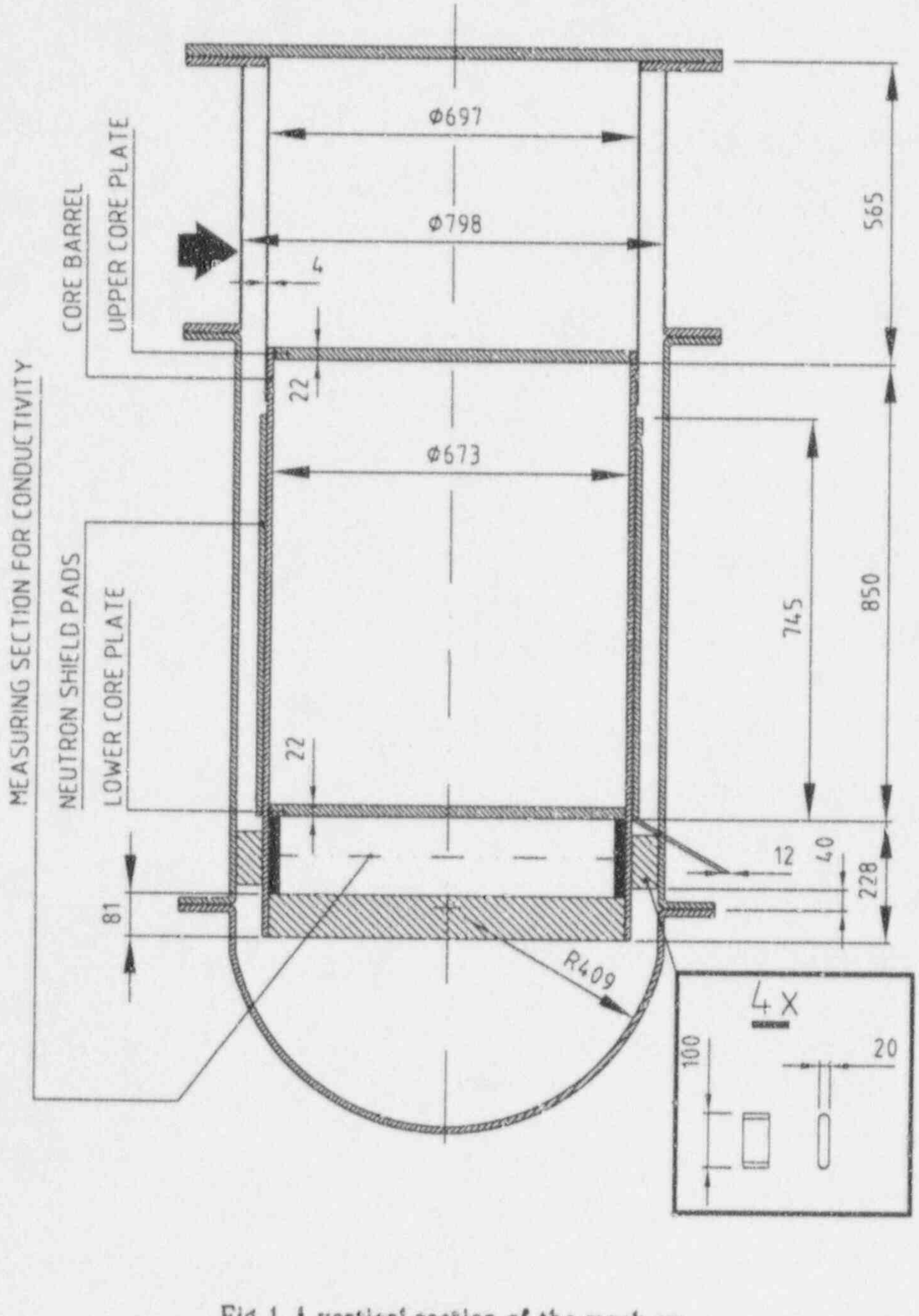


Fig. 1 A vertical section of the mock-up.

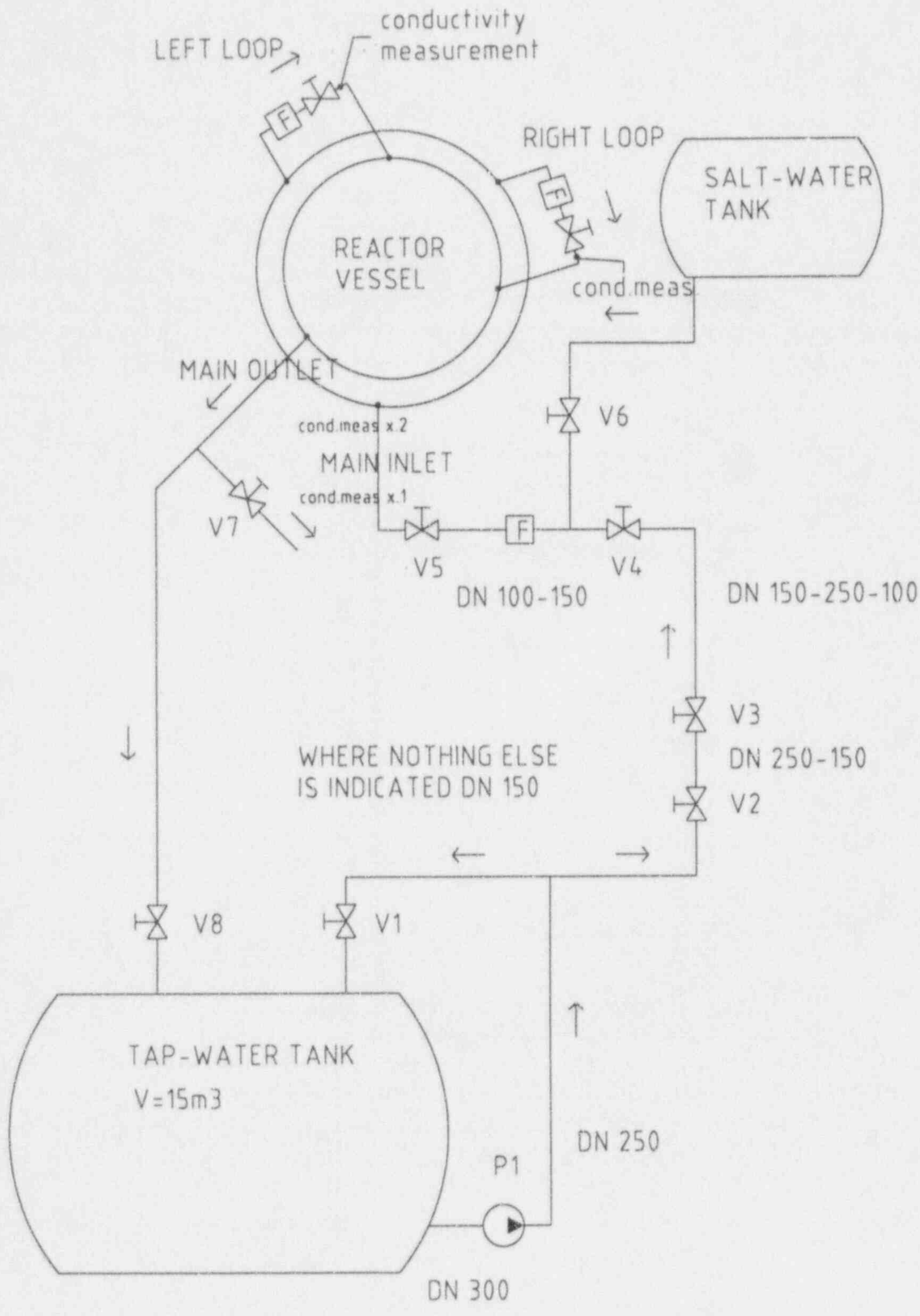


Fig. 2 A sketch of the experimental set-up.

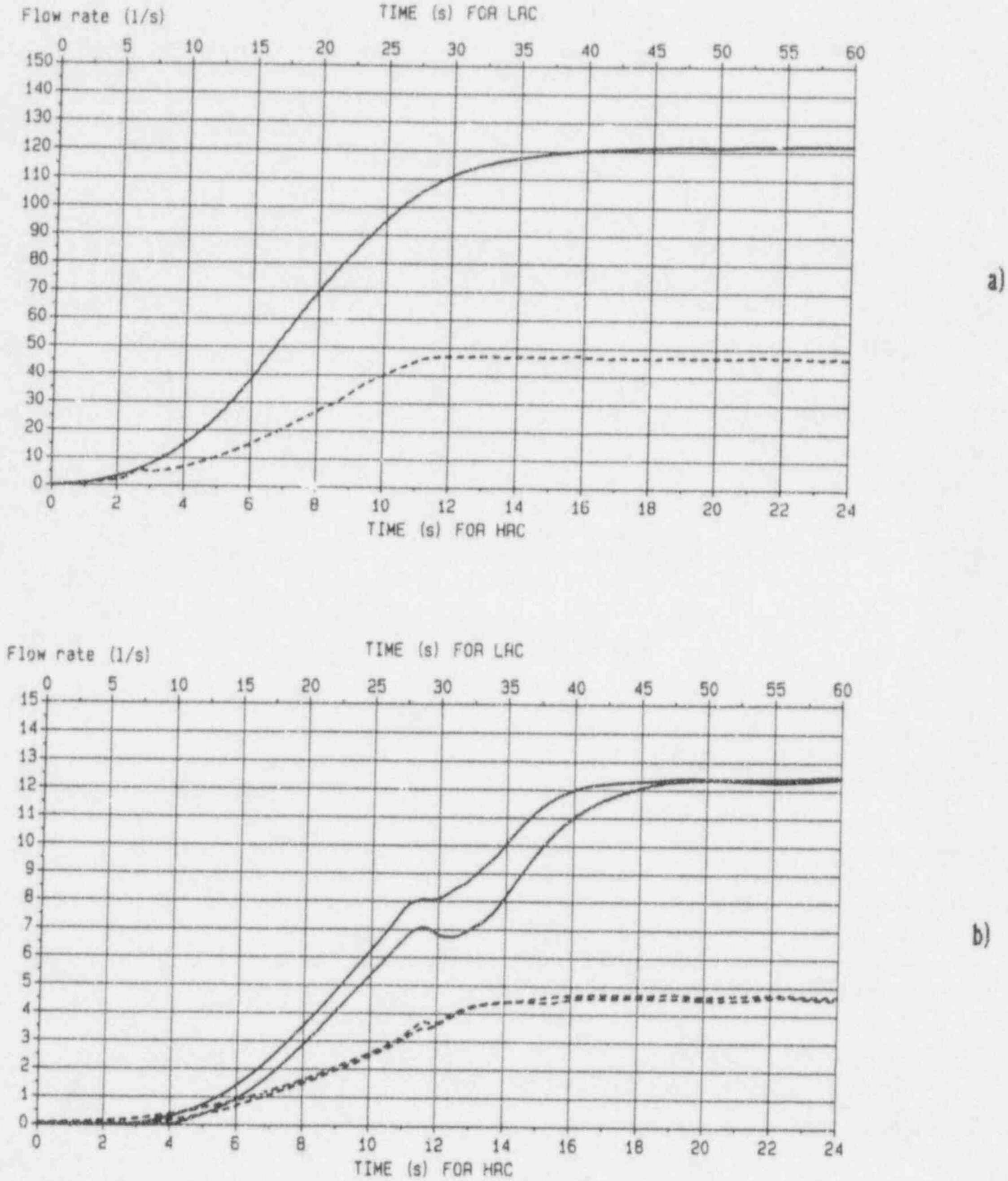
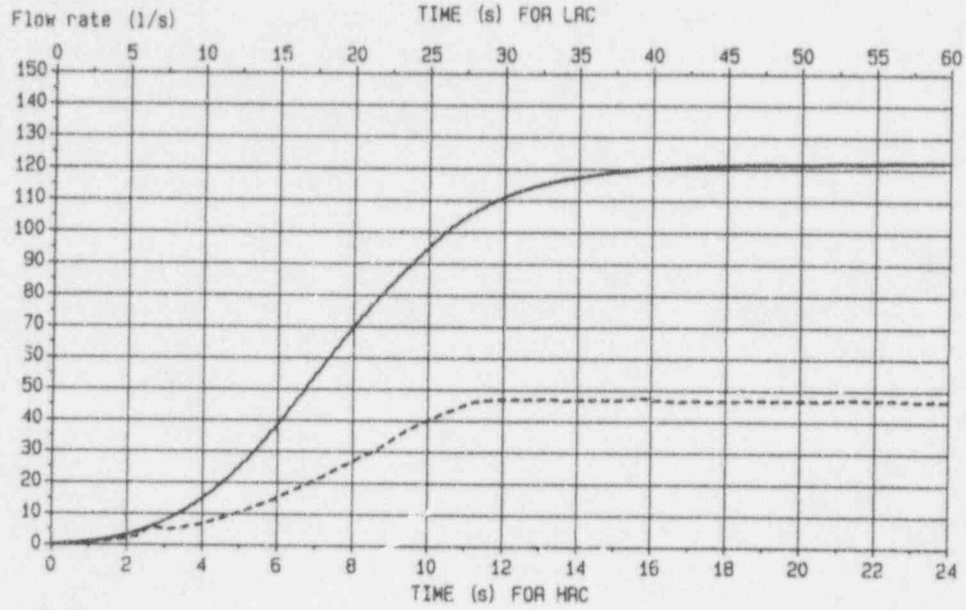
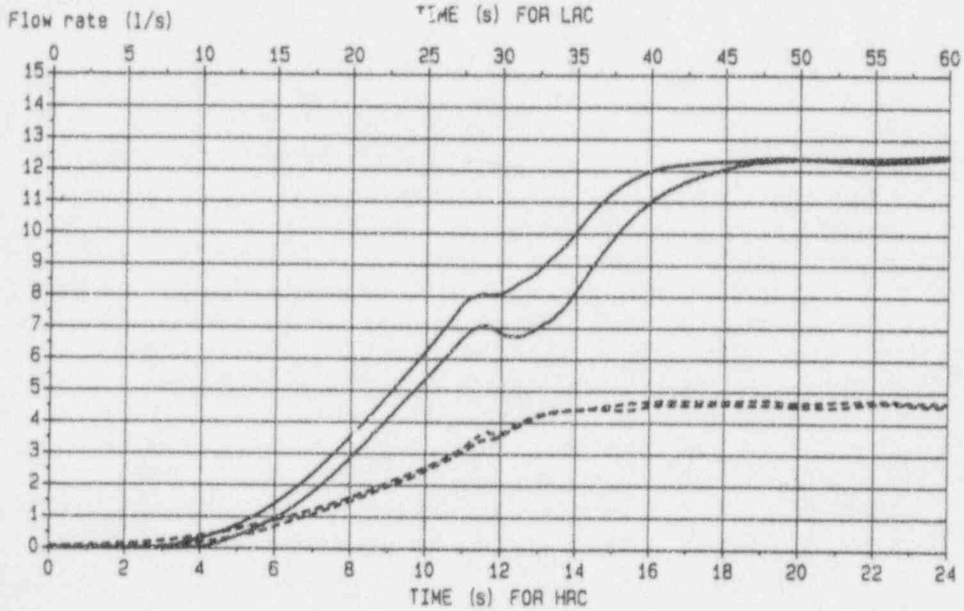


Fig. 3 Temporal variations of the flow rate through the a) main inlet and b) side loops (----------; LRC, __________; HRC).



a)



b)

Fig. 3 Temporal variations of the flow rate through the a) main inlet and b) side loops (----------; LRC, __________: HRC).

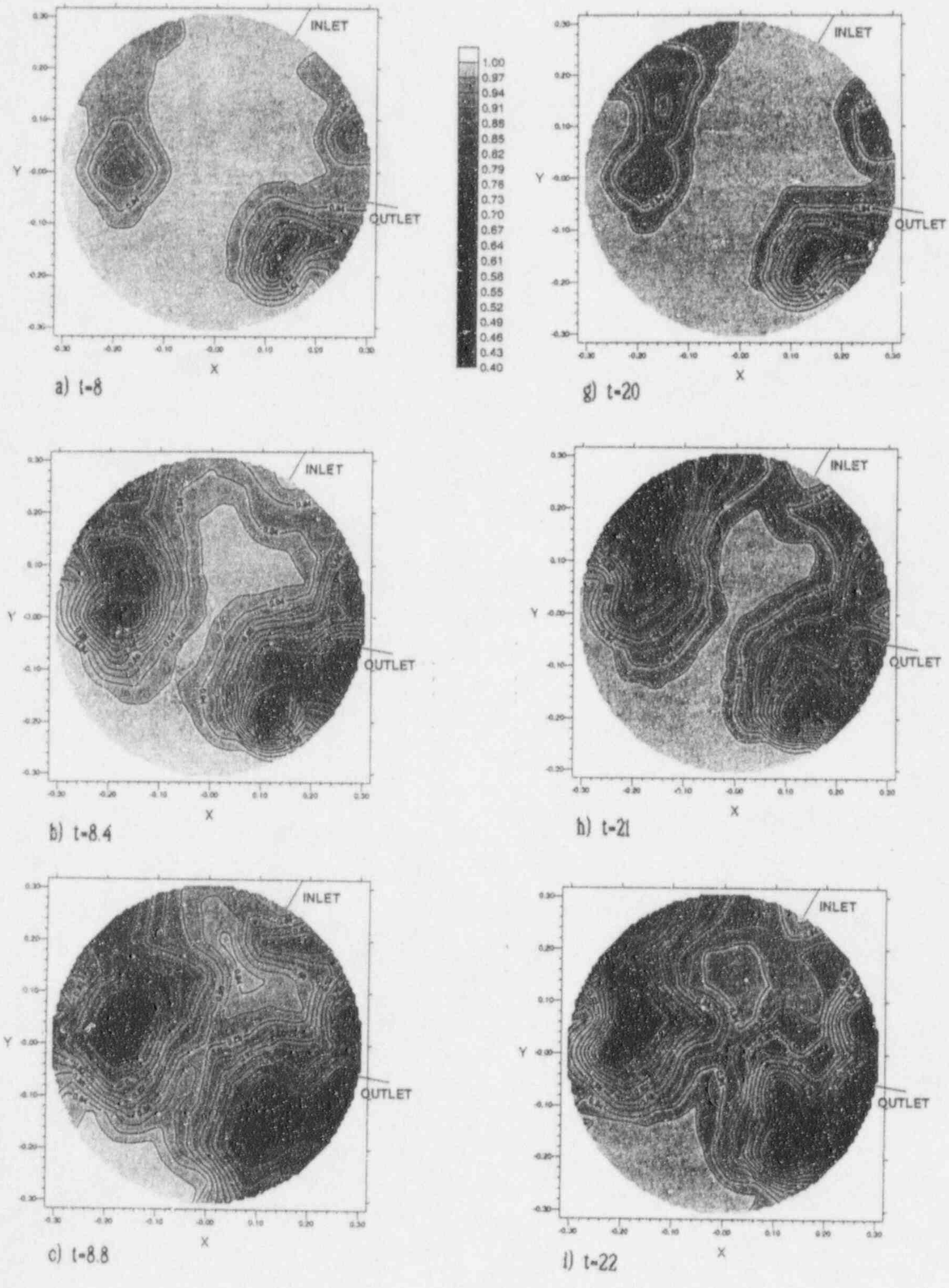


Fig. 5 Nondimensional concentration at the core inlet and for different values of time. The left column is the HRC and the right column is the LRC.

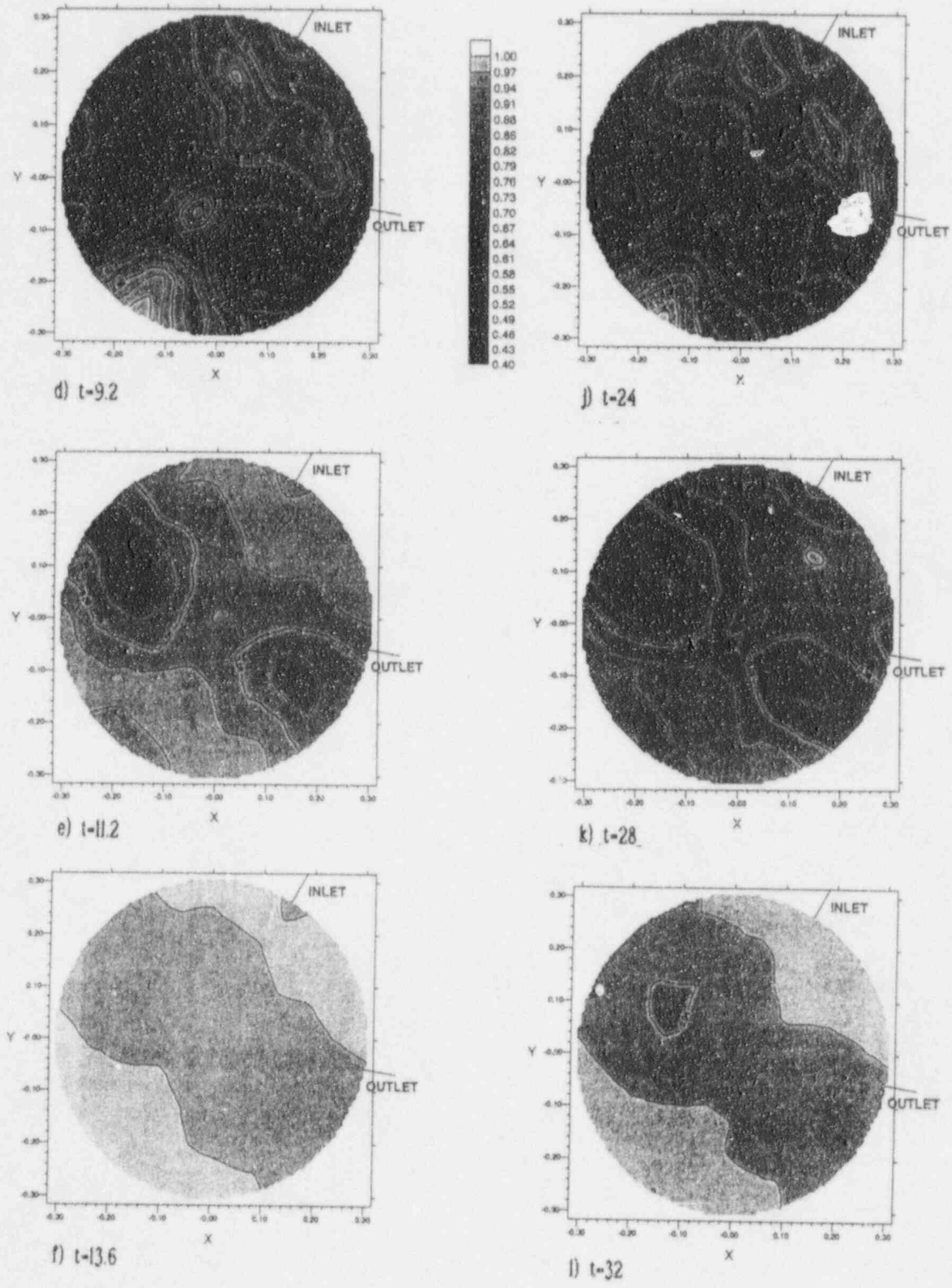
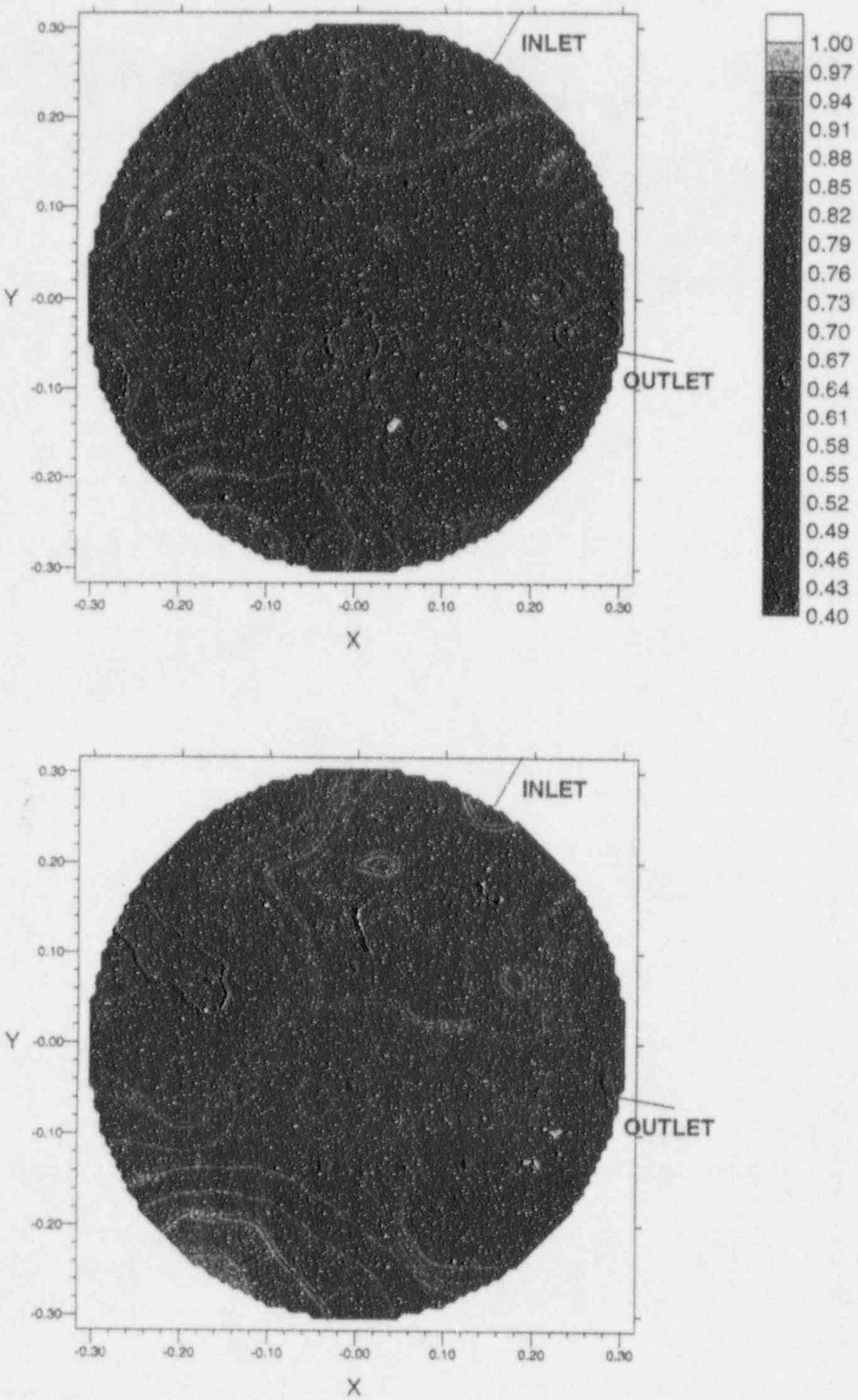


Fig. 5 continued.



a)

b)

Fig. 6 Ensemble-averaged minimum nondimensional concentration at the core inlet. a) HRC and b) LRC.

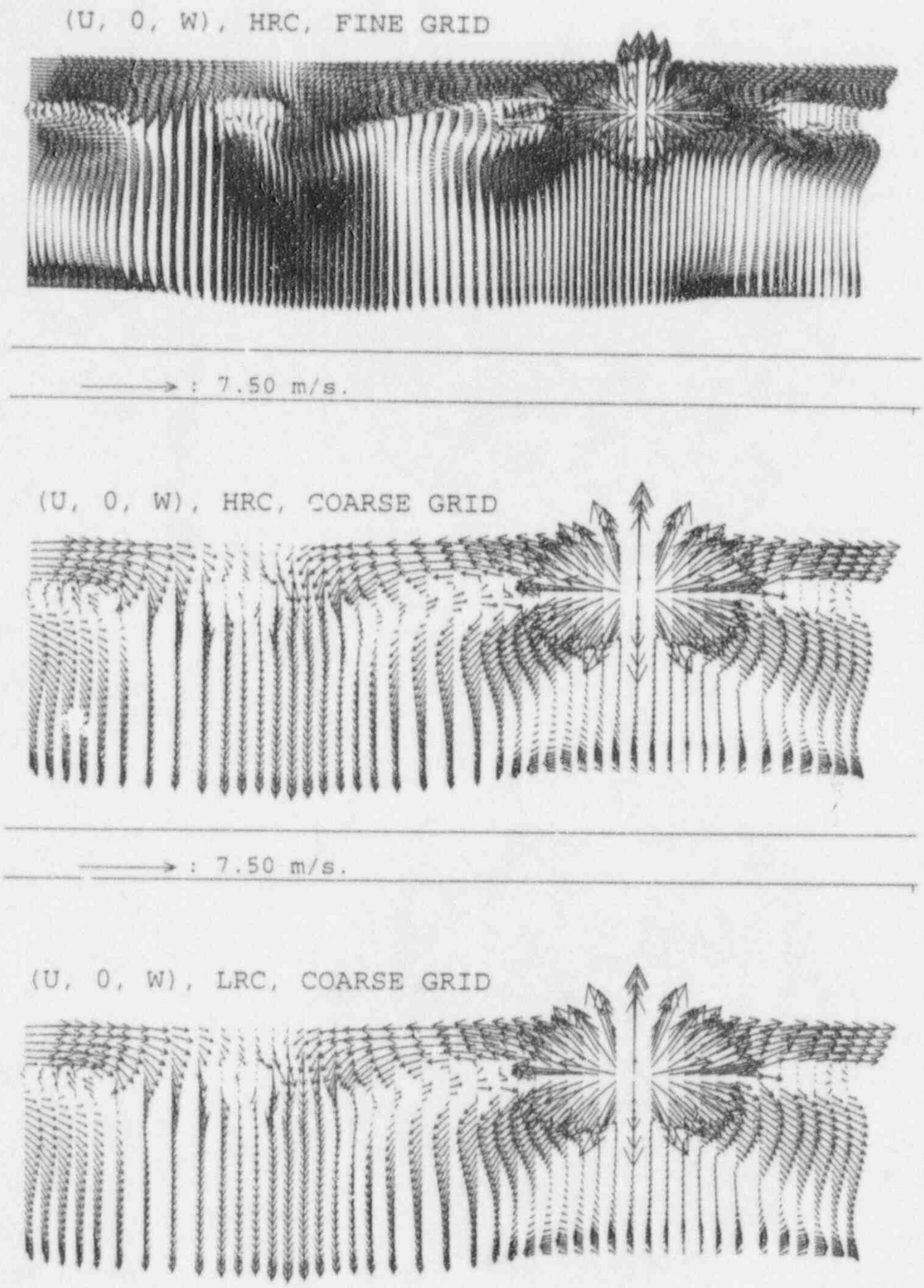


Fig. 7 Computed velocity vectors in a vertical plane in the middle of the downcomer. (The radial component of velocity is turned off in these figures.)

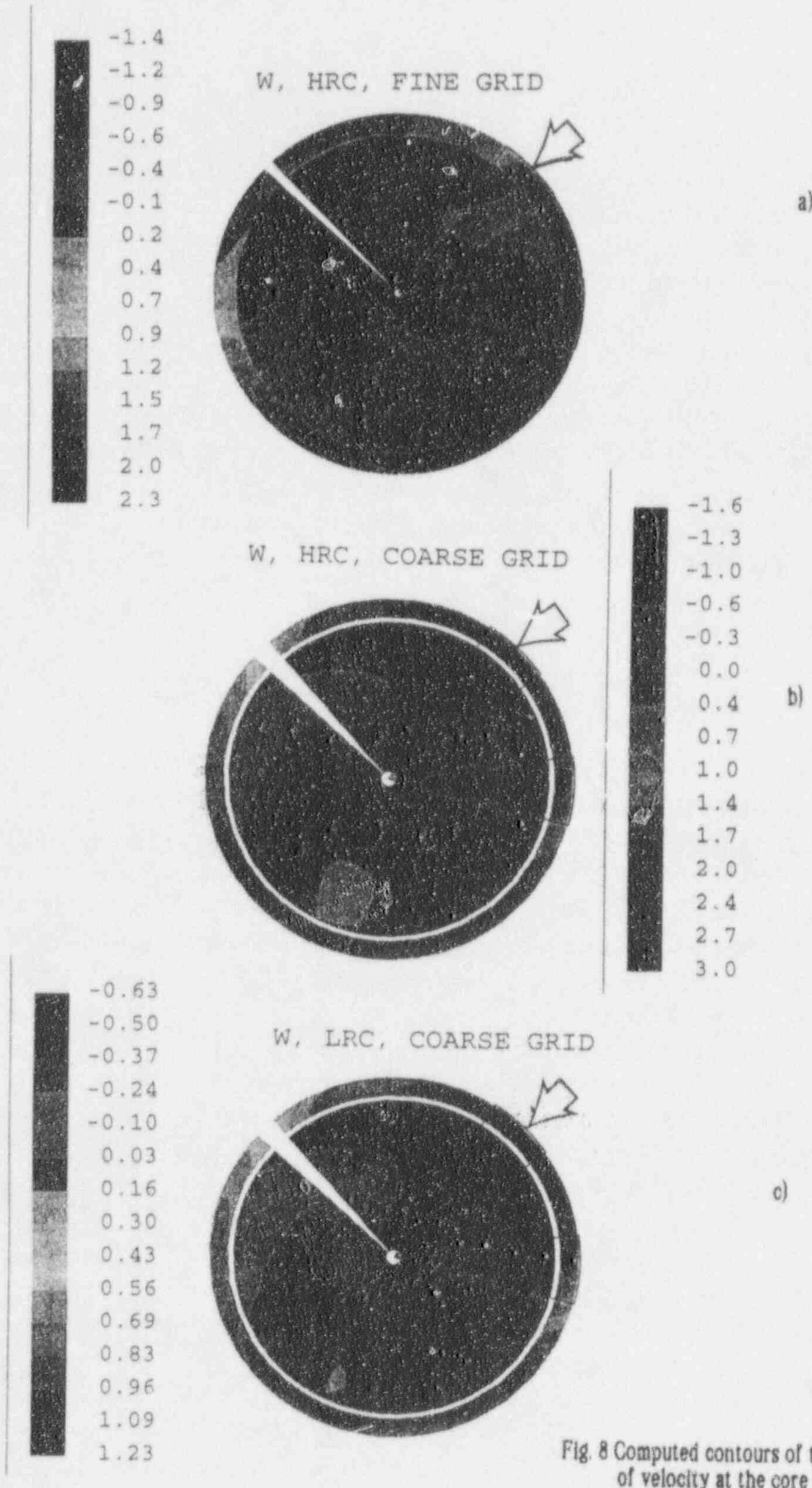


Fig. 8 Computed contours of the vertical component of velocity at the core inlet. a) fine grid. b) and c) coarse grid. The arrows show the positions of the main inlet and outlet.

NRC FORM 335 (2-89) NRCM 1102, 3201, 3202 — U.S. NUCLEAR REGULATORY COMMISSION

BIBLIOGRAPHIC DATA SHEET

(See instructions on the reverse)

1. REPORT NUMBER (Assigned by NRC, Add Vol., Supp., Rev., and Addendum Numbers, if any.)

NUREG/CP-0158
NEA/CSNI/R(96)3

2. TITLE AND SUBTITLE

Proceedings of the OECD/CSNI Specialists Meeting on Boron Dilution Reactivity Transients

Held in
State College, Pennsylvania, USA
October 18-20, 1995

3. DATE REPORT PUBLISHED

MONTH	YEAR
June	1997

4. FIN OR GRANT NUMBER

L1971

5. AUTHOR(S)

N/A

6. TYPE OF REPORT

Conference Proceeding

7. PERIOD COVERED *(Inclusive Dates)*

8. PERFORMING ORGANIZATION - NAME AND ADDRESS *(If NRC, provide Division, Office or Region, U.S. Nuclear Regulatory Commission, and mailing address; if contractor, provide name and mailing address.)*

Organized by
OECD Nuclear Energy Agency
The Pennsylvania State University

9. SPONSORING ORGANIZATION - NAME AND ADDRESS *(If NRC, type "Same as above"; if contractor, provide NRC Division, Office or Region, U.S. Nuclear Regulatory Commission, and mailing address.)*

Division of Systems Technology
Office of Nuclear Regulatory Research
U.S. Nuclear Regulatory Commission
Washington, DC 20555-0001

10. SUPPLEMENTARY NOTES

D. Ebert, NRC Project Manager

11. ABSTRACT *(200 words or less)*

A CSNI Specialist Meeting on Boron Dilution Reactivity Transients was held in State College, Pennsylvania, USA, from October 18-20, 1995. The meeting was sponsored by the United States Nuclear Regulatory Commission (USNRC) in collaboration with the Committee on the Safety of Nuclear Installation (CSNI) of the OECD Nuclear Energy Agency (NEA) and the Pennsylvania State University. The objective of the meeting was to bring together experts involved in the different activities related to boron dilution transients, to promote discussion among these experts, and to focus on the technical issues of concern in resolving the safety significance of such events.

12. KEY WORDS/DESCRIPTORS *(List words or phrases that will assist researchers in locating the report.)*

Boron Dilution, Reactivity Initiated Accidents, Boron Mixing, Reactivity Transients, Thermal-Hydraulic Analysis, Boron Transport, Computational Fluid Dynamics

13. AVAILABILITY STATEMENT

unlimited

14. SECURITY CLASSIFICATION

(This Page) unclassified

(This Report) unclassified

15. NUMBER OF PAGES

16. PRICE

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, DC 20555-0001

OFFICIAL BUSINESS
PENALTY FOR PRIVATE USE, $300

SPECIAL STANDARD MAIL
POSTAGE AND FEES PAID
USNRC
PERMIT NO. G-67

120555139531 1 1AN1R11R419L
US NRC-CIRM
PUBLICATIONS BRANCH
TPS-PDR-NUREG
2WFN-6E7
WASHINGTON DC 20555