Proceedings of the
SPECIALIST MEETING
ON SEVERE ACCIDENT
MANAGEMENT IMPLEMENTATION

Organised by
OECD NUCLEAR ENERGY AGENCY
in collaboration with
NORTHEAST UTILITIES

Niantic, Connecticut, USA
12-14 June 1995

COMMITTEE ON THE SAFETY OF NUCLEAR INSTALLATIONS
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NUCLEAR ENERGY AGENCY

STEERING COMMITTEE FOR NUCLEAR ENERGY

COMMITTEE ON THE SAFETY OF NUCLEAR INSTALLATIONS

SPECIALIST MEETING ON SEVERE ACCIDENT MANAGEMENT IMPLEMENTATION

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Organized in Collaboration with Northeast Utilities

This meeting is organized by CSNI's Senior Group of Experts on Severe Accident Management (SESAM)
OECD

The Convention establishing the Organization for Economic Cooperation and Development (OECD) was signed on December 14, 1960.

Pursuant to Article 1 of the Convention, the OECD shall promote policies designed:

- to achieve the highest sustainable economic growth and employment and a rising standard of living in Member countries, while maintaining financial stability, and thus to contribute to the development of the world economy;

- to contribute to sound economic expansion in Member as well as nonmember countries in the process of economic development; and

- to contribute to the expansion of world trade on a multilateral, nondiscriminatory basis in accordance with international obligations.

The current Signatories of the Convention are Australia, Austria, Belgium, Canada, Denmark, Finland, France, the Federal Republic of Germany, Greece, Iceland, Ireland, Italy, Japan, Luxembourg, Mexico, the Netherlands, New Zealand, Norway, Portugal, Spain, Sweden, Switzerland, Turkey, the United Kingdom, and the United States. The Commission of the European Communities takes part in the work of the OECD (Article 13 of the OECD Convention).

NEA

The OECD Nuclear Energy Agency (NEA) was established on February 1, 1958, under the name of the OEEC European Nuclear Energy Agency. NEA membership today consists of all European member countries of OECD as well as Australia, Canada, Japan, Republic of Korea, Mexico, and the United States. The Commission of the European Communities takes part in the work of the Agency.

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

NEA works in close collaboration with the International Atomic Energy Agency (IAEA), with which it has concluded a Cooperation Agreement, as well as with other international organizations in the nuclear field.
CSNI

The NEA Committee on the Safety of Nuclear Installations (CSNI) is an international committee made up of scientists and engineers. It was set up in 1973 to develop and coordinate the activities of the OECD Nuclear Energy Agency concerning the technical aspects of the design, construction, and operation of nuclear installations insofar as they affect the safety of such installations. The Committee's purpose is to foster international cooperation in nuclear safety amongst the OECD Member countries.
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OECD SPECIALIST MEETING ON
SEVERE ACCIDENT MANAGEMENT IMPLEMENTATION
(Niantic, Connecticut, USA; June 12-14, 1995)

PROGRAM

GENERAL CHAIRMAN: MARIO V. BONACA (NU)

SUNDAY, JUNE 11, 1995

18:00 WELCOMING RECEPTION AND REGISTRATION - HOTEL

MONDAY, JUNE 12, 1995

7:30 - 8:30 REGISTRATION

8:30 - 10:00 WELCOME:
J. F. Opekta, Executive Vice President, Nuclear - NU
M. V. Bonaca, Program Chairman

INTRODUCTORY SESSION
Per Bystedt - Chairman, SESAM
Gary Holahan, USNRC
W. H. Raisin/D. Modeen, NEI

10:00 BREAK

10:30 SESSION I - "Approaches to Severe Accident Management Program Development"
Chairman: Dr. H. Tuomisto (IVO, Finland)

10:30 - 11:00 "Update on the Technical Basis for the Severe Accident Management Guidelines," by B. Chexal, A. Singh and J. Chao (EPRI, USA), and R. E. Henry (FAI, USA)


11:30 - 12:00 "BWR Owners' Group Accident Management Guidance," by F. A. Emerson (BWROG, USA)

12:00 - 13:00 LUNCH
13:00 - 13:30  "Japanese Regulatory Position on Severe Accident Management and Related Research Activities," by T. Fujishiro and J. Sugimoto (JAERI) and M. Hirano (NUPEC/INS, Japan)

13:30 - 14:00  "Severe Accident Management: Presentation of and Technical Basis for the Current EDF Approach," by M. Vidard (EDF, France)

14:00 - 14:30  "Severe Accident Management Simplified," by R. J. Lutz, Jr., D. K. Ohkawa, J. T. Taylor, and B. S. Monty (Westinghouse Electric Corp., USA)

14:30 - 15:00  "Severe Accident Management Development Program and Insights for Closure of the Industry Accident Management Process," by William G. Dove, Mark. A. Greene, (ABB-CE Nuclear Operation, USA), and Yehia F. Khalil (NU, USA)

15:00 - 15:30  BREAK

15:30 - 16:00  "Accident Management Measures and Strategies in Germany—An Overview," by E. Kersting, U. Erven, and B. Putter (GRS, Germany)

16:00 - 16:30  "Swedish Regulatory Aspects on Severe Accident Management Implementation," by O. Sandervag and W. Frid (SKI, Sweden)


17:00 - 17:30  GENERAL DISCUSSION

19:00  DINNER CRUISE

TUESDAY, JUNE 13, 1995

8:00  SESSION II - "Severe Accident Management Implementation"

Chairman:  Nigel E. Buttery
            (Nuclear Electric, UK)
8:00 - 8:30  "EDF's Experience in the Implementation of Severe Accident Management Provisions," by G. Serviere (EDF, France)

8:30 - 9:00  "Accident Management Implementation at Carolina Power & Light Company," by F. A. Emerson (CP&L, USA)

9:00 - 9:30  "Development of Accident Management Measures in GKN II," by E. Grauf (GKN II, Germany)

9:30 - 10:00  "Severe Accident Management Strategies for PWR Plants in Japan," by K. Shigemune, K. Yoshihara and M. Ohtani (KEPCO, Japan)

10:00 - 10:30  "Severe Accident Management at Northeast Utilities," by M. L. Van Haltern (NU, USA)

10:30 - 11:00  BREAK

11:00 - 12:00  SIMULATOR TOUR

12:00 - 13:00  LUNCH

13:00 - 13:30  "Accident Management for BWR Plants in Japan," by, Y. Tomioka T. Zama, K. Miyata and A. Omoto (TEPCO, Japan)

13:30 - 14:00  "Report of the Nuclear Power Station Gundremmingen (KRB II)," by W. Stadelmann (KGB, Germany)

14:00 - 14:30  "Severe Accident Management: Implementation Activities at Duquesne Light's Beaver Valley Power Station," by R. K. Brosi (Duquesne Light Company, USA)

14:30 - 15:00  "Provisions for Emergency Situations in the Krümmel Nuclear Power Plant," by W. Stubbe and U. Welte (KK GmbH, Germany)

15:00 - 15:30  BREAK


16:00 - 16:30  "Severe Accident Management at the Forsmark, OKG, and Barsebäck NPP:S," by V. Gustavsson (Forsmark), H. Dubik (OKG), P. Jacobsson (Baseebäck)
16:30 - 17:00

"Concept of Severe Accident Management for NPP-91 Design with VVER-1000 Reactor," by G. A. Antropoov (Russia)

"Concept of Severe Accident Management for NPP-91 Design with VVER-1000 Reactor," by H. A. Antropov (Atomenergoaproekt, Russia)

17:00 - 17:30
GENERAL DISCUSSION

17:30
SITE TOUR

WEDNESDAY, JUNE 14, 1995

8:00 a.m. - 12:30 p.m.  SESSION III - "Uncertainties and Open Issues"

Chairman: J. Rohde (GRS, Germany)

8:00 - 8:30
"General introductory paper on remaining issues, uncertainties and open items of severe accident management," by J. Rohde (GRS, Germany)

8:30 - 9:00
"Overview: Uncertainties remaining in severe accident phenomenology," by R. E. Henry (FAI, USA)

9:00 - 9:30
"Reactor Cavity Flooding as an Accident Management Strategy," by I. Catton and W. E. Kastenberg (UCLA, USA)

9:30 - 10:00
BREAK

10:00 - 10:30

10:30 - 11:00
"Perspectives on the Utility Severe Accident Management Guidelines," by L. W. Ward, D. J. Hanson, and D. Brownson (INEL, USA)

11:00 - 11:30
"In Pursuit of Consistency and Completeness in the Severe Accident Assessment and Management," by H. Tuomisto (IVO, Finland)
11:30 - 12:00
"Severe Accident Management at Ringhals PWR - Present Status and Future Work," by G. Löwenhielm and S. Jacobsson (Vattenfall, Sweden)

12:00 - 13:00
LUNCH

13:00 - 14:00
"Lithuanian Views and Activities in the Area of Severe Accident Management Implementation," by Evaldas Bubelis (LEI, Lithuania)

"Implementation of Severe Accident Measures in the Netherlands," by George Vayssier (Nuclear Safety Dept., The Netherlands)

"KRSKO NPP Individual Plant Examination (IPE) Insights as a Knowledge Source for Accident Management Development," by M. Gregoric, and D. Vojnovic (SNSA, Slovenia)

General Discussion - "General Panel Discussion on Severe Accident Management Implementation Issues and Closing Remarks"
SUMMARY AND CONCLUSIONS OF THE SPECIALIST MEETING
Mario V. Bonaca, SAMI Specialist Meeting Program Committee Chairman, Welcomes Attendees and Presents the Schedule of Events.
SUMMARY AND CONCLUSIONS OF THE SPECIALIST MEETING

OECD SPECIALIST MEETING ON
SEVERE ACCIDENT MANAGEMENT IMPLEMENTATION

The OECD Specialist Meeting on Severe Accident Management (SAM) Implementation was held in Niantic, Connecticut, U.S.A., June 12 - 14, 1995. It was hosted by Northeast Utilities at the Training Simulator Building located at the Millstone Nuclear Station. About sixty (60) experts from fourteen (14) OECD member countries attended the meeting, as well as experts from Lithuania, Russia and Slovenia. Thirty-three (33) papers were presented in three (3) sessions.

The purpose of this meeting was to bring together experts involved in the implementation of severe accident management. These experts came from research organizations, regulatory authorities, vendors and utilities. It is worthwhile to note that about one-half of the participants represented utilities. The meeting had, therefore, a very practical and concrete tone. Information was shared and discussed on the knowledge needed and the problems encountered during the implementation phase of all the provisions necessary to successfully manage severe accidents: guidelines, hardware modifications, training and organizational aspects.

The meeting enabled the Senior Group of Experts on Severe Accident Management (SESAM) of OECD's Nuclear Energy Agency to gather information needed to complete a report for the Committee on the Safety of Nuclear Installations (CSNI) documenting severe accident management implementation in OECD countries.

This is the second and final report on the status of SAM programs in OECD countries that SESAM prepares for CSNI. The first was published in 1992, and focused on the development of SAM programs in OECD countries. A Specialist Meeting on SAM Program Development was held in Rome, Italy in September of 1991 to help gather information SESAM needed to write the first report. Since significant progress towards SAM program implementation had been noted in all OECD member countries since 1991, CSNI requested SESAM to produce a second and final report documenting SAM program implementation in member countries.

The Niantic Specialist Meeting was structured around three main themes, one for each session of the meeting.

During the first session, papers from regulators, research groups, designers/owners groups and some utilities discussed the critical decisions in SAM, how these decisions were addressed and implemented in generic SAM guidelines, what equipment and instrumentation was used, what are the differences in national approaches, etc.

During the second session, papers were presented by utility specialists that described approaches chosen for specific implementation of the generic guidelines, the difficulties encountered in the implementation process and the perceived likelihood of success of their SAM program in dealing with severe accidents.

The third and final session was dedicated to discussing what are the remaining uncertainties and open questions in SAM. Experts from several OECD countries presented significant perspectives on remaining open issues.
SAM GUIDELINES IMPLEMENTATION AND MAINTENANCE

There was general consensus that major progress toward SAM guidelines implementation has been made. Extensive efforts have gone into developing comprehensive and robust guidelines that have been, or are being used, for the development of plant specific SAM guidelines or procedures. These are valuable tools that the utilities should implement as soon as practical.

There was general recognition that although sufficient information is available to allow SAM implementation to proceed, all issues are not closed yet, and new information will become available that may affect the content of the current guidelines. Utilities should be encouraged to maintain and enhance the effectiveness of their SAM program by performing periodic reviews and updates to incorporate significant new information. These periodic updates should not require extensive effort, because all new information can be screened to assess if it has an impact on existing guidelines. It was suggested by some participants that the industry should explore means of performing such screening at the Owners' Group or research facility level for all plants.

A concluding perspective presented at the Specialist Meeting was the need of feedback from implementation to research. This is because translation of phenomena insights into SAM guidelines has been a complex process, and it is not always clear that all strategies can be implemented, or that implemented strategies can be effective.

REMAINING UNCERTAINTIES IN SAM

In summary, there was general agreement that much progress has been made, but some uncertainties still remain and some additional work needs to be done.

There was no general agreement on the extent of uncertainties still remaining to be addressed. Expert opinions ranged from a perspective that the only remaining uncertainties which may impact SAM measures had to do with in-vessel and ex-vessel debris coolability, to other perspectives that included, among remaining significant uncertainties, other issues or phenomena, such as long-term containment pressurization, possible re-criticality of corium debris, coolability of pressure vessel, corium-steel mixing and its impact on reactor pressure vessel integrity.

In part, this variation of opinions is due to the varying degree of knowledge that individual SAM programs require to support SAM actions. Since SAM dedicated equipment and actions vary from plant to plant and among national programs, a varying degree of knowledge of possible phenomena is required to support SAM decisions; thus, varying perspectives exist on remaining uncertainties.

There was a general recognition of four significant issues regarding uncertainties:

1) Disagreement regarding remaining uncertainties is probably much less than it appears on the surface, if the stated uncertainty is complemented by answering the question: "Will further knowledge to reduce this uncertainty change the currently recommended operator action?" For example, all participants agreed that "water is good": In absence of further information, water should be fed in all cases to a molten core. There is also general agreement that there are uncertainties regarding the effectiveness and possible downsides of such action in some situations. Some experts believe that some degree of uncertainty will always remain, such that further work to reduce this uncertainty will not change the recommended operator action to add water: therefore, they consider this issue significantly closed. Others believe that additional work can and should be done, that could refine or modify the recommended operator action to add water in some situations.
2) Some uncertainties, even large, will always exist in SAM. Regarding further work to be done to reduce such uncertainties, the question must be asked: "will it change the currently recommended operator action?" This operator action should be one of the focuses of such research.

3) Meanwhile, operators should be provided SAM guidelines that are direct and support the action that they need to take, even if there are possible negative outcomes of that action. SAM training should recognize the downsides of the action, so that the operators are aware of the potential negative consequences, but guidelines should reinforce the action to be taken.

4) One of the focuses of continuing severe accident R&D relates to the information required to optimize proposed ALWR design features, rather than being essential for SAM decision making.

DIFFERENCE IN SAM PHILOSOPHY OF NATIONAL APPROACHES

At the national level, due to differences in approaches to economic issues and societal risks, practical implementation has been made in a wide variety of regulatory contexts and technical options. From a regulatory standpoint, situations range from voluntary actions made on the basis of agreements between the utilities and the regulatory bodies, to the imposition of requirements that severe accidents safety goals be met together with release limits. This varied range of situations has resulted in significant differences in the amount of hardware modifications included into the SAM programs in different countries. These differences in hardware dedicated to the management of severe accidents also influence the perceived need for further work to address uncertainties in areas where additional information is needed to operate such equipment.

However, it was recognized that when design and operation of existing plants provide adequate protection of public health and safety, enhancements to address SAM must be balanced with costs and the need to maintain the economic viability of the nuclear option.

ORGANIZATIONAL ISSUES

The issue of organizational efficiency in case of complex malfunctions or severe accidents was also dealt with and there was consensus on the following:

- Responsibilities and the chain of command must be clearly defined and known to all staff.
- The added complexity resulting from AM guidelines will require training, periodic exercises and monitoring to assure effectiveness of the Emergency Plan.
- Possible drawbacks or deficiencies which could appear during such exercises need to be analyzed and corrected accordingly.

Differences were observed among national programs, generic guidelines and specific utilities in the assignment of responsibilities for SAM guidelines implementation during SAM. The differences relate to the respective roles and responsibilities of control room and technical support staff in their shared effort to manage Severe Accidents. However, in general, the responsibility for all actions taken at the plant remains with plant personnel.
CONCLUSIONS

1) The meeting was very successful and achieved its intended purposes. Major progress toward SAM guidelines implementation has been made in recent years. The SESAM group gathered valuable information for its report to CSNI. A clear picture of the state of implementation was obtained, and the reasons behind the differences were made clear.

2) Open issues and remaining uncertainties were identified. However, they are not of such importance as to impede implementation. In addressing these remaining issues, their impact on the SAM guidelines has to be taken into account.

3) The translation of phenomena insights into SAM guidelines has been a complex process. It is not clear that all strategies can be implemented, nor that implemented strategies can be always effective. New questions might arise during the implementation process. Therefore, there should be a feedback process from implementation to research.

4) Low power and shutdown states were not covered in detail during the meeting. Severe accidents during these states have specific characteristics and pose specific challenges. Clear picture of how this could impact SAM strategies was not provided in the meeting. Further work on this topic is recommended.
SESSION I

APPROACHES TO SEVERE ACCIDENT MANAGEMENT PROGRAM DEVELOPMENT

The session consisted of thirteen (13) papers from seven (7) countries and concentrated on the development of severe accident management (SAM) strategies and resulting generic guidelines and their bases.

A definition of SAM was provided early in the session and repeatedly referred to during the meeting, showing a general consensus of all participants with this definition:

"SAM consists of those actions that are taken by the plant staff during the course of an accident to:

- Prevent core damage
- Terminate progress of core damage and retain the core within the vessel
- Maintain containment integrity
- Minimize offsite releases

SAM also involves pre-planning and preparatory measures for:

- SAM guidance and procedures
- Equipment modifications to facilitate procedure implementation
- Severe accident training

The overall objective is to further reduce the risks of large releases. It is the responsibility of the licensees to develop and implement a SAM program."

This definition includes the concept that there is some overlap between what is referred to as Accident Management (AM) and Severe Accident Management (SAM).

From the presentations provided during this session, it is apparent that the development of SAM programs in different countries are highly influenced by the general expectations set at the national level for such programs.

In some countries, risk reduction through SAM programs is pursued by simply applying existing equipment and instrumentation when developing SAM guidelines and procedures. Minor equipment modifications for SAM are made whenever they are cost-effective in facilitating the plant staff to apply procedures. Major plant modifications have been implemented over the past several years but were generally focused on prevention of core damage, rather than management of a damaged core in vessel or in containment.
In other countries instead, SAM is considered a basis of design by requiring that certain severe accident safety goals and release limits have to be met. This approach can lead to major plant modifications that are needed for ensuring a SAM safety goal.

Some other countries have chosen to combine features of both of these approaches.

Development of SAM guidelines and procedures differ from country to country depending on the number and diversity of Nuclear Power Plants and on the number of utilities.

In the U.S.A., the technical basis for SAM guidelines and procedures was developed by EPRI. The Owners' groups used the EPRI work to develop generic SAM guidelines for different plant types. The Severe Accident Issue Closure Guidelines issued by NEI provides an approach to bring closure to SAM issues with an industry initiative. As a final step, the utilities have started to prepare the plant-specific procedures and guidance based on these documents. The approach taken in implementing SAM procedures and guidance will be licensee-specific according to different plant design features, technical support and operational capabilities of the utilities.


In France, EdF has developed its SAM approach in close cooperation with safety authorities and Framatome by optimum use of existing systems, implementing a number of modifications on plant and implementing specific procedures.

In Canada, Ontario Hydro is in the early stages of developing SAM guidelines for its CANDU plants.

In Germany, the licensees have agreed to implement SAM measures on a voluntary basis after recommendation of RSK in 1986. The SAM procedures and guidance cope mainly with the beyond design basis range, and are an extension of safety-function-oriented procedures.

In Sweden, the SAM implementation was completed in 1988 for all plants. Since that time, the efforts of the Swedish Industry have concentrated on maintaining and upgrading their SAM capability.

In Korea, KAERI has started to develop a research program to resolve plant-specific SAM issues.

In the Netherlands, the licensees have been requested to re-baseline and backfit their plants to modern safety standards. As part of this process, also SAM hardware will be added; SAM procedures will then be developed and integrated with the symptom-based procedures already in place.
SESSION II

SEVERE ACCIDENT MANAGEMENT IMPLEMENTATION

The session consisted of fifteen (15) papers from nine (9) countries and clearly illustrated that great progress has been made since the Rome meeting (1991). There were many common elements in the strategies, and, in general, the overall objective was the same: to build upon the strengths of existing emergency arrangements. There were also differences in the strategies adopted and in the way they were implemented, which were largely the result of plant design differences and differences in national regulatory or operational arrangements.

The presentations came from countries at different stages of implementation ranging from Sweden, who had achieved full implementation in 1988, to the U.S.A. and Japan, who have developed generic guidance and are programmed to complete implementation towards the end of the 1990s. The papers, therefore, provided feedback or experience from each phase of implementation. The common themes of the papers are summarized below:

(I) Development of Guidance and Procedures

In the development of their accident management guidance and procedures, most utilities are making use of the insights gained from PSAs/IPEs to focus efforts. Though useful, this is not an essential element of a program; for instance, Sweden and France started developing their original procedures before the availability of a plant specific PSA. The development of severe accident management has been a voluntary effort by the utilities in most countries, with the support and encouragement of their regulatory authorities. In the Scandinavian countries, quantitative regulatory targets have been specified.

In most cases, accident management has been applied to already operating plants and so must build on and be consistent with existing operating procedures. Examples were presented of cases where accident management was considered in the design conception phase (VVER 91), in the detailed design stage (Sizewell B), and during construction (GKN II).

There was general agreement that severe accident management procedures should avoid the need for changes to existing procedures. They should try to avoid contradictions with such procedures since this could introduce confusion into the decision making process. If it is necessary to take actions which are inconsistent with normal practices, the instructions need to be clear and simple. Over-sophisticated procedures should be avoided.

The papers indicated a widespread use of Critical Safety Functions, symptom-based approaches in guidance and procedures.

(II) Severe Accident Management Measures

There were differences in emphasis in the approaches adopted to the provision of measures aimed at severe accident mitigation. Everyone seems to make full use of existing equipment but some utilities have also implemented hardware changes. On existing plants, many backfits and improvements have been made to enhance preventive measures. On new plants, such measures form part of the base design. For additional accident management provisions, the need for cost effectiveness was emphasized.
The measures implemented for which there was a measure of agreement in terms of basic objectives are:

- the addition of water for core/debris cooling.
- cooling the containment.
- provision of back-up power supplies.

There was more diversity in terms of the means to achieve the objectives. For instance, water injection can be achieved by the use of ECCS, fire water systems or mobile equipment. The containment can be cooled by sprays, containment coolers, external sprays (for steel containments) or even by venting. Containment coolers can be cooled by any source of water or ultimate heat sinks. Backup power supplies can be provided by additional diesels (fixed or mobile) or by connection to other offsite power sources. To make maximum use of existing equipment, the means to override interlocks has been provided in some cases.

Areas where there are still differences in approach include hydrogen control (for large dry containments) and the use of filtered venting.

(iii) Organizational Implementation and Training

Severe accident management measures must be implemented in the framework of the existing emergency organization and build on this foundation. The need to involve and make use of additional technical support staff is recognized but there were apparent differences in the roles and responsibilities with respect to Severe Accident Management Guidance in different utilities. The plant knowledge and experience of the Control Room Operators was generally recognized but many utilities favor transferring to, or sharing with the TSC or crisis team, responsibility for decision making for SAMGs, so as not to overburden the operator. Although specific SAMG documentation is provided to TSC and Crisis Team staff, control remains within the plant. In general, the emergency organization works as part of a team with ultimate responsibility resting with the utility emergency organization. In Canada, the decision to vent rests with the Provincial Authorities. In papers from countries such as France who have experience of exercises using their accident management procedures, the role of good teamwork and communications between the teams involved was emphasized. Some changes to the structure of the emergency organization had taken place as a result of this to better integrate the safety engineer with the operational team and to avoid the crisis teams taking over on-site responsibilities.

Many of the apparent differences appeared to be associated with differences of approach as far as external emergency arrangements and regulations were concerned.

The need to train people at all levels in SAMG, including basic knowledge of physical phenomena, was recognized though concerns about the burden this may place on operators was expressed. In general, it was felt that all those involved should understand why actions were being taken.

The use of drills and exercises had proved valuable for those utilities that had already implemented their SAMGs and those in the process of implementation instead to use these as an integral part of the update/refinement process as well as for training purposes. The need to make the SAMGs "living documents" and to update them was generally agreed.
SESSION III

UNCERTAINTIES AND OPEN ISSUES

In this session, six (6) presentations were given, concentrating on remaining uncertainties in severe accident phenomenology and trying to give a picture of the resolution status with respect to accident management decision making. Furthermore, unresolved issues, particularly in the area of the phenomenological behavior that can introduce uncertainties into some of the decisions, were discussed to better understand potential side effects of implementing some of the accident management strategies.

For the Swedish and Finnish approaches, the consistency and completeness of accident management measures together with important issues remaining were presented and discussed. One specific paper dealt with investigations on how to design adequate computerized tools and information systems for assistance in accident management.

There was a general understanding that uncertainties in severe accident phenomenology will always remain, but that the knowledge base exists and has been used to develop plant specific solutions. It is important to note that the sources of uncertainties can be identified and have to be taken into account when developing specific severe accident measures. In cases where these uncertainties remain unacceptably high, so that the effectiveness of a specific measure becomes questionable or the measure could lead to serious adverse effects, an alternate strategy should be considered when possible and decisions made accordingly.

The basic approach to develop a plant specific SAM concept requires deterministic investigation. Sources of uncertainties could be the:

- experimental data base for specific severe accident phenomena
- application of experimental results for real plant conditions
- physical and mathematical modeling techniques in computer codes
- limited experience of the code user in modeling technical systems
- measured parameters that could be affected by accident conditions

But also the use of PSA - work to support the development of a SAM concept contains sources of uncertainties. All such potential sources of uncertainties have to be carefully investigated, to see clearly the influence on the accident management concept being developed.

One main point of discussion concentrated on strategies to keep molten core material inside the reactor pressure vessel by late phase water injection or early cavity flooding for cooling the vessel outside. Additional research work seems to be necessary to understand and realize the different heat transfer modes involved and the mechanism creating a gap between reactor pressure vessel bottom head and lower crust of the molten pool, providing the space for water ingestion. If such strategies to prevent vessel breach could be demonstrated to be effective for different reactor designs, there was a general consensus that this will drastically reduce the importance of several uncertainties, especially for those phenomena dominating the ex-vessel melt behavior. As an
INTRODUCTORY SESSION
Per Bystedt, Chairman of SESAM, Welcomes the Conference Participants
Ladies and gentlemen,

1. Severe accident management is the subject of this specialist meeting. Severe accident management for the purpose of making nuclear power plant operation more safe. I am convinced that everyone present in this room agrees that a sound accident management programme can reduce risk from nuclear power significantly. We all know about incidents that actually have happened and in which the use of

   - a sound strategy and good instructions
   - and of suitable instrumentation and other equipment
   - and through an effective communication among the operators in the plant and between them and specialists outside the plant

have showed to be effective in the course of termination of the incident. We all have read about such incidents and studied them carefully for the purpose of learning and giving feedback to our development. Some of us have experience from simulated incidents in drills and in full-scale simulators and some of you might even have hands-on experience from a control room or a response centre from real incidents. I do not foresee any debate or questioning at this meeting of the need for accident management. Nor of the fact that the benefits from accident management lower the probability for incidents to happen; that is because working out and implementing the programme can reveal weaknesses in safety which can be removed before they might initiate an event. However, the main benefit is to provide a better preparedness and higher confidence for the involved personnel to handle the plant, if difficult situations should occur. The preparedness they should have is to stabilize the plant with a minimum of radiological consequences to the personnel and to the environment and a minimum of damage to the plant. A severe accident programme is a necessary component to give public confidence in nuclear power.

2. It is good to see that so many organisations from countries all over the world are being represented at the meeting. It is also good to see that countries from Eastern Europe that previously have not shared so much of the work within the CSNI are represented. This gives us an excellent opportunity to learn about experience from many different organizations with different characteristics. It is important to remember that there is no such thing as one universally optimal severe accident management programme. This is because a program consists of an integration of technical factors and organisational factors. It is a matter of integration of duties of the personnel at the plant, at central utility management and technical support and at local and central authority organisations. Each country and even each power plant must find its optimal programme based on the structure and on the traditions specific for each case. Certainly there are many common factors that many plants can share; we will for sure discuss many such common factors at this meeting, but at the end it is the responsibility of each and every plant, in communication with the authorities, to work out and to implement its own programme filling its own needs.

3. The real key word to this specific meeting is the word 'implementation'. An accident management programme might be designed and worked out very nicely in all details but might still not contribute very much to safety. Only a programme that has the required content and that has been effectively implemented at the power plant and in the organisations involved will have a chance to be effective the day it will be called upon. Compromises have to be made in the programme in order to make it work practically within the limitations of available human and technical resources, organisational and human factor aspects, costs etc. Programme implementation is therefore a key link for safety with respect to accident management.

4. There are variations in the stages of implementation of severe accident management in different plants throughout the world for reasons such as differences in priorities, in organisational and regulatory systems
and many more. Even so, I think it is important for all people involved in accident management, despite the stage, to bear implementation in mind. From the early development stage when attention is paid to forming the programme and to preparing the people that will be involved all the way until late stages for an existing programme where the important matter is to secure that the effectiveness of the programme does not decline with time due to insufficient attention. As for all safety matters, periodic checking is necessary to keep the information up to date. New knowledge from research and new experience from e.g. exercises that might be at hand must be assessed for the purpose of confirmation and necessary updating of strategies, procedures and tools included in the programme. Most certainly training of new operators and other personnel must be organised as well as retraining of the personnel must be done periodically.

5. The SESAM group bears the responsibility for initiating and for organizing this meeting; forgive me, I am certainly not forgetting the special programme committee headed by Mario Bonaca that have put together the programme, done all the paper reviews and arranged all the practical preparations. SESAM has worked since 1980 with information exchange and with processing of technical information and advice to CSNI in the area of Severe Accident Management. The participation in the group is a mix of delegates from ten countries; mainly people with experience from research and regulation together with representatives for US, Japanese and European utilities which are mainly people representing technical development, operation and management. This gives an optimal mix of knowledge and experience that has encouraged a very constructive and balanced discussion in the group over the years. The group has previously organized two specialist meetings; one in Rome in 1991 on Severe Accident Programme Development and one in Halden 1993 on Operator Aids for Severe Accident Management and Training. One major report on Severe Accident Management; Prevention and Mitigation has previously been issued. Several responses to CSNI have been given on specific requests.

6. The meeting held in Rome in 1991, which many of us attended together, was in fact the starting point for the planning of this meeting. We then found that there is a common understanding among the CSNI countries of the basic structures and major elements in severe accident management. We also found the variations in the implementation stages and that a lot of work was in progress. This meeting that we now are entering gives us the opportunity to report on further progress in development and on further experience from implementation in our programmes.

For SESAM, the meeting also has a very special purpose. The present task for SESAM is to report to CSNI on the subject of severe accident management implementation in September this year. The meeting will therefore help us to monitor the status of the work and the important questions that are being put forward and to learn about your experience from processing these questions. This information will help us provide quality to the report to CSNI.

The following questions are of interest:

- How did you define the critical decision points in your severe accident management strategy?
- How did you technically support your choice of actions at those points?
- What are the main issues in your programme that need further attention?
- What are your main experiences from implementation of your programme?

SESAM will listen carefully to the presentations and to the discussions to extract such information.

This SESAM's "agetic" purpose is not argument enough to invite you all to Niantic. Far from it. The main purpose is because we are convinced that you sharing experience with each other in severe accident management will be mutually beneficial for your future work with these matters. And by that, the probability of an accident in the future having severe consequences will be further reduced. That is the utmost goal of our work. I am convinced that the time and resources we all have allocated to this meeting are being well spent for the purpose of risk reduction.

I am looking forward to an interesting meeting and a nice stay in Niantic.

Thank you!
ACCIDENT MANAGEMENT IN THE USA

UNITED STATES NUCLEAR REGULATORY COMMISSION

GARY M. HOLAHAN, DIRECTOR
DIVISION OF SYSTEMS SAFETY AND ANALYSIS
OFFICE OF NUCLEAR REACTOR REGULATION
U.S. NUCLEAR REGULATORY COMMISSION
Use by Utilities
Products (Guidance/Methods/Materials) for
NRC Since 1998 has led to Development of
Extensive Interactions Between Industry and

A Voluntary Industry Effort
of Clear Requirement, NRC has Pursued A/M as
In View of Industry Support for A/M and Lack

May 1998)
Closure of Severe Accident Issues (SECY-88-147,
Element of the NRC Integration Plan for
Accident Management (A/M) is an Essential

OVERVIEW
ACCIDENT MANAGEMENT: A DEFINITION

• Actions Taken by the Plant Staff During the Course of an Accident to:
  - Prevent Core Damage
  - Terminate Progress of Core Damage and Retain the Core Within the Vessel
  - Maintain Containment Integrity
  - Minimize Offsite Releases

• Involves Pre-Planning and Preparatory Measures:
  - Accident Management Guidance and Procedures
  - Minor Equipment Modifications to Facilitate Procedure Implementation
  - Severe Accident Training
Managers in the Procedures and Guidance

- Training Operators, Technical Support Staff, and
  - Operating Procedures and Guidance
  - Preparing and Implementing Severe Accident
  - Evaluating Information on Severe Accidents

Management Plan Which Provides a Framework For:

To Have Each NRC Licensee Implement an Accident

A/M PROGRAM
FUNDAMENTAL OBJECTIVE OF
EXPECTED UTILITY STEPS IN IMPLEMENTING A/M

1. Implement Industry Products

2. Institutionalize the A/M Plan to Maintain Capabilities and Accept New Information Should it Become Available

3. Perform Periodic A/M Drills and Self-Assessments
Severe Accident Training Materials/Lesson Plans

- Methodology/Criteria for Assigning A/M Strategies to Either Control Room or TSC (BWRs)
- Severe Accident-Related EPG Changes (BWRs)
- Severe Accident Management Guidelines (SAMG)
- Scope and Content of Utility A/M Effort
- Formal Industry Position Statement of Expected

IMPLEMENTATION

MAJOR PRODUCTS FOR
NRC PLANS FOR MONITORING IMPLEMENTATION

• Limited Number of Sites to be Visited to Inspect Plant-Specific Implementation. Emphasis on:
  - Interface of SAMG with EOPs and Emergency Plan, and
  - Incorporation of SA Material into Personnel Training

• Inspections Most Efficiently Performed Through Observation of SAMG Application During EP Exercise or A/M Drill

• Further Dialog Planned With Industry/NEI to Define Specific Process for Confirming Adequate Implementation
(Revisions to Examiner Standards, Workshops, and Examination on Severe Accident Material)

- Clarity Guidance to Inspectors Regarding Training Implementation Details
- Continue to Work with Industry on Methodology

- Review BWROG A/M Products When Submitted
- Needed
- Modifications to Other Inspefections Modules as Temporary Instruction for A/M Audits

- Inspection Guidance as Appropriate for Evaluating A/M Implementation, and Develop
- Continue Dialogue with Industry on Specific Process

- Dissemination to Commission and Regions
- Program and Expectations/Perceptions for A/M
- Develop Comprehensive Description of A/M

REMAINING NRC ACTIONS
**SUMMARY**

- A/M Program will Enhance Capabilities to Prevent and to Mitigate Severe Accidents

- Closure will be Achieved on a Plant-by-Plant Basis
  - On a Voluntary Basis, in Accordance with Industry Initiative
  - Using Industry- and NRC-Developed Guidance and Methods

- A/M Capabilities will be Maintained "Living"
  - Periodically Exercised by Conducting Utility A/M Drills
  - Periodically Updated by Utility to Incorporate New Information

- NRC will maintain Oversight of A/M to Assure Program Effectiveness
ABSTRACT

U.S. NUCLEAR INDUSTRY FORMAL POSITION ON SEVERE ACCIDENT MANAGEMENT

BY

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TO

CSNI Specialists Meeting on Severe Accident Management Implementation
Waterford, Connecticut
June 1995

In its simplest form, severe accident management is defined as "a set of practical actions a nuclear plant staff can take to prevent, control or mitigate the consequences of a severe core damage accident." In parallel with each plant's conduct of the Individual Plant Examinations for Internal and External Events (IPE and IPEEE), generic severe accident management technical and programmatic guidance was being developed by the U.S. nuclear industry for individual plant implementation. The purpose of this paper is to summarize plant-specific actions being taken by U.S. nuclear utilities over the next four years.

On November 4, 1994, the Chief Nuclear Officers of U.S. nuclear utilities approved a formal position on severe accident management requiring action at each plant. It states that:

*Each licensee will:

Assess current capabilities to respond to severe accident conditions using Section 5 of NEI 91-04, Revision 1, "Severe Accident Issue Closure Guidelines."

Implement appropriate improvements identified in the assessment, within the constraints of existing personnel and hardware, on a schedule to be determined by each licensee and communicated to the NRC, but in any event no later than December 31, 1998.*
The goal of severe accident management is to enhance the capabilities of the Emergency Response Organization (ERO) to establish core cooling and ensure that any current or immediate threats to the fission product barriers are being managed. Post-TMI actions and IPE insights have already addressed most aspects of preventing core damage. The focus of the industry effort is to provide guidance where Emergency Operating Procedures (EOPs) are no longer effective, or revise EOPs if appropriate, and make full use of existing plant capabilities, including standard and non-standard uses of plant systems and equipment.

Significant interaction among utility, INPO, EPRI, Nuclear Steam Supply System Owners Groups, NRC, and other recognized experts produced the foundation of actions and plant response from which plant-specific severe accident management guidance can be developed. These actions can be categorically divided into elements similar to those described by the NRC in SECYs 88-147 and 89-012:

- Severe accident management guidance/strategies
- Severe accident training
- Computational aids for technical support
- Informed needed to respond
- Delineation of decision-making responsibilities
- Utility self-evaluation

To achieve closure, each licensee is expected to perform the following steps:

- Evaluate industry-developed bases and Owners Group severe accident management guidance (SAMG) along with the plant IPE, IPEEE and current capabilities, to develop severe accident management guidance for accidents found to be important in their plant. Other generic and plant-specific information (e.g., NRC and industry studies, PSA results, etc.) will be considered, as appropriate;
- Interface SAMG with the plant’s Emergency Plan;
- Incorporate severe accident material into appropriate training programs; and
- Establish a means to consider and possibly adopt new severe accident information from licensee self assessments, applicable NRC generic communications, PRA studies, etc.

In summary, licensee-specific actions will include developing plant-specific guidance and training from generic material, interfacing the enhanced severe accident capabilities with existing procedures and organizations, and establishing self assessment capabilities.

Because this is an industry initiative, there are no specific regulatory criteria. Rather, industry has defined its goals and objectives by its actions relative to severe accident management.
U.S. NUCLEAR INDUSTRY
FORMAL POSITION ON
SEVERE ACCIDENT MANAGEMENT

by
David J. Modeen
Nuclear Energy Institute

OECD Specialist Meeting on
Severe Accident Management
Implementation
Niantic, CT
June 12, 1995
PRESENTATION TOPICS

- Formal Industry Position
- Expected Utility Actions
- Regulatory Oversight
- Conclusions
FORMAL INDUSTRY POSITION

- The Chief Nuclear Officers of U.S. utilities approved the following action:

"Each licensee will:

- Assess current capabilities to respond to severe accident conditions using Section 5 of NEI 91-04, Revision 1, "Severe Accident Issue Closure Guidelines."

- Implement appropriate improvements identified in the assessment, within the constraints of existing personnel and hardware, on a schedule to be determined by each licensee and communicated to the NRC, but in any event no later than December 31, 1998."
PRIMARY MOTIVATION AND PERSPECTIVE RELATIVE TO FORMAL INDUSTRY POSITION

- SAM is capstone of the IPE and IPEEE; capture pragmatic, operational insights

- Need to resolve severe accident issue in order to make progress in risk- and performance-based regulation

- SAMG packages represent sound, well developed guidance.

- Desire a conscientious, but focused effort (no “gold plating”).

- Creation of a specific program is not intended, but in fact discouraged

- Recognize NRC has a role, but with an industry initiative it is limited
EXPECTED UTILITY ACTIONS
RESOURCE IMPLICATIONS

• Use plant personnel, including:
  – Emergency Planning
  – Operations
  – Training
  – Engineering
  – Senior Management

• Integrate SAM into existing Emergency Response Organization (ERO) functions in the:
  – Control room
  – Technical Support Center (TSC)
  – Operations Support Center (OSC)

• Changes depend on current ERO staffing & structure
EXPECTED UTILITY ACTIONS
SAMG DEVELOPMENT & IMPLEMENTATION

- Assess vendor-specific SAMG documents along with plant-specific insights (IPE and IPEEE), information sources, and computational capabilities to develop SAMG

- Integrate SAMG into the plant's Emergency Plan and associated implementing instructions

- Incorporate severe accident material into appropriate training material

- Establish means to consider and possibly adopt new severe accident information
EXPECTED UTILITY ACTIONS
APPLY SYSTEMS APPROACH
TO TRAINING

• Analysis
  – Determine need (new guidance), job (ERO positions), tasks (SAMG structure)

• Design
  – Define objectives and performance measures (e.g., mini-drills)
  – Selection of instructional strategies (classroom vs. self-study)

• Development (lesson plans)
  – Utility responsibility based on NSSS-specific and INPO materials

• Implementation
  – Depends on qualifications of individuals assigned to ERO position

• Evaluation
EXPECTED UTILITY ACTIONS
TRAINING - PRIMARY
CONSIDERATIONS

• Highest priority remains accident prevention; don’t detract from effectiveness of existing training programs

• Current full scope simulators are designed to support realistic and effective training for normal, abnormal, and emergency procedures - no plan to further enhance simulator capabilities

• Commensurate with severe accident job responsibilities
  – Evaluators (TSC)
  – Decision-makers (TSC)
  – Implementers (Control Room & OSC)
EXPECTED UTILITY ACTIONS
TRAINING - PRIMARY
CONSIDERATIONS

- Operators are most knowledgeable about alternative equipment alignments
- Operators must understand the basis for TSC directions
  - Core damage factors
  - Critical components
  - Potential strategies
  - Severe accident phenomenology
- Training must emphasize those aspects of SAMG that go against EOP training
- Estimate up to 16-24 hours of initial training for an evaluator on INPO and owners group lesson material; less for others
EXPECTED UTILITY ACTIONS
SELF EVALUATION

• Initial (following implementation of formal position)
  – Ensure feasibility and usefulness
  – Integration without adverse effect on emergency response

• Ongoing (subsequent)
  – Table-top or inter-facility mini-drills. Should be separate from E-Plan exercises to allow “free-play”
  – Periodic, with critiques to capture lessons learned and creative or innovative preventive and mitigative measures
  – Intended to train on, evaluate, and improve in-plant SAM response capability
REGULATORY OVERSIGHT
BREAKING THE PARADIGM

• No NRC Safety Evaluation Report on SAMG or other industry products

• Utilities implement under § 50.59

• No utility submittals other than estimated completion date

• Rely on utility self-evaluation using typical EP-type critique practices

• Conduct limited number of site visits to assess plant-specific implementation

• Audit findings would not result in EP "weakness" determination unless overall effectiveness is decreased
REGULATORY OVERSIGHT
REMAINING DISCUSSION TOPICS

• Complete review of BWROG Overview and EPG/SAG documents

• Develop NRC temporary instruction for site implementation visits

• Schedule and conduct implementation visits; identify lessons learned and provide feedback to industry

• Determine long term NRC role in assessing severe accident management capabilities

• Document role of regional NRC staff
CONCLUSIONS

- Common basis and philosophy among U.S. utilities regarding severe accident management is complete

- Plant-specific knowledge needed to assess generic materials is complete

- Uncertainties remain and always will

- Therefore,

  MAKE IT HAPPEN!!
SESSION I

APPROACHES TO SEVERE ACCIDENT MANAGEMENT

PROGRAM DEVELOPMENT
SESAM MEMBERS

Back Row (from left): Mario Bonaca, Jurgen Rohde, Nigel Buttery, Per Bystedt, John Lehner, Benoit De Boeck, and Gianfranco Capponi.

Front Row (from left): Harri Tuomisto, Toshio Fujishiro, and Jacques Royen.

Not in Photo: Thomas King, Michel Vidard, and Peter Wigfull.
UPDATE ON THE TECHNICAL BASIS FOR THE SEVERE ACCIDENT MANAGEMENT GUIDELINES

By:

Bindi Chexal (EPRI), Avtar Singh (EPRI),
Jason Chao (EPRI), and Robert Henry (FAI)

Presented at the Specialist Meeting on Severe Accident Management Implementation
Niantic, Connecticut
June 12-14, 1995

ABSTRACT

EPRI sponsored the development of the technical basis for the Severe Accident Management Guidelines. These were completed and distributed to the members of the Joint Owners Group (WOG) in 1992. Each of the four owners groups within the United States used this technical basis to develop a design specific approach to severe accident management. These owners' group guidelines are being used to formulate plant specific guidance which effectively mesh with the Emergency Operating Procedures (EOPs).

Since 1992 significant additional studies have been performed through industry sponsored work as well as work sponsored by the Department of Energy and the Nuclear Regulatory Commission at the national laboratories that should be recognized and incorporated into the technical basis. In many cases, this additional work further solidifies the technical basis initially used by the owners groups and in one case raises an issue related to how containment flooding may be considered in the SAMGs. This paper presents an update on the technical basis and briefly discusses how this new information modifies, if at all, or reinforces the previous conclusions.

1. INTRODUCTION

EPRI sponsored the development of a technical basis report to be used as the foundation for developing severe accident management guidelines (SAMGs) (Henry, 1992). In this two-volume report, the major phenomenological issues were (a) defined, (b) examined with respect to the anticipated severe accident behavior given the state-of-the-art for the individual phenomenon
and (c) used to assess their influence on possible candidate high-level actions that could be taken to recover from an accident. These were investigated with respect to three RCS damage conditions:

- OX - cladding OXidation,
- BD - Badly Damaged core, and
- EX - significant core debris is accumulated in the containment (Ex-Vessel).

Furthermore, four descriptors were used for potential containment damage states and these include:

- CC - Containment is Closed (isolation is complete) and heat removal is available,
- CH - containment isolation is complete but the containment integrity could be CHallenged,
- I - containment Isolation function is not complete, and,
- B - containment is Bypassed, i.e. RCS isolation function is not complete or may be challenged as a result of the accident sequence.

As a result of the work performed over the last three years, the candidate high-level actions have remained the same (see Table 1), the characterization in terms of potential core damage and containment damage states has remained the same and to a large extent the phenomenological assessments remain unchanged. To a large extent, the phenomenological evaluations presented in the TBR remain unchanged. However, in a few areas, namely external cooling of the RPV, direct containment heating, ex-vessel coolability and a new area, in-vessel cooling, there have been significant developments that should be acknowledged when developing plant-specific guidance. In many cases, these additional experiments and analyses further solidify the directions given in 1992 and in the case of in-vessel cooling, present a perspective of an additional issue, which is consistent with the observed behavior in the TMI-2 accident. Because of its importance in potentially stopping the accident progression and protecting the RPV integrity, this issue should be considered when evaluating the conditions for containment flooding. These additional developments are the subject of this paper.
Table 1
Candidate High Level Actions
and Special Considerations

Candidate High Level Actions

1. Inject into (Makeup to) RPV/RCS
2. Depressurize the RPV/RCS
3. Spray Within the RPV (BWR)
4. Restart RCPs (PWR)
5. Depressurize Steam Generators (PWR)
6. Inject into (Feed) the Steam Generators (PWR)
7. Spray into Containment
8. Inject into Containment
9. Operate Fan Coolers
10. Operate Recombiners
11. Operate Igniters
12. Inert Containment with Noncondensables (BWR)
13. Vent Containment
14. Spray Secondary Containment
15. Flood Secondary Containment

Special Considerations

1. External Cooling of RPV/RCS
2. Steam Inerting of the Containment
2.0 CURRENT STATUS OF THE PHENOMENOLOGICAL EVALUATIONS

Volume two of the TBR is devoted to the discussion of "the physics of accident progression." Table 2 is taken from the Table of Contents for this volume and illustrates how the phenomenological issues associated with severe accidents were organized to support the development of severe accident management guidelines. Each of the subject areas listed is a
### Table 2

**STATUS OF THE PHYSICS OF ACCIDENT PROGRESSION**

<table>
<thead>
<tr>
<th>1.0</th>
<th>CORE</th>
<th></th>
</tr>
</thead>
<tbody>
<tr>
<td>A</td>
<td>Decay Heat</td>
<td>NC*</td>
</tr>
<tr>
<td>B</td>
<td>Core Uncovery</td>
<td>NC</td>
</tr>
<tr>
<td>C</td>
<td>Secondary Side Dryout</td>
<td>NC</td>
</tr>
<tr>
<td>D</td>
<td>Cladding Behavior (Collapse, Ballooning, Embrittlement, Oxidation/Hydrogen Generation)</td>
<td>NC</td>
</tr>
<tr>
<td>E</td>
<td>Fuel Behavior (Melting, Eutectics, Liquification)</td>
<td>NC</td>
</tr>
<tr>
<td>F</td>
<td>Core Geometry (Shattering, Columnar Buckling, Debris Bed Formation)</td>
<td>NC</td>
</tr>
</tbody>
</table>

<table>
<thead>
<tr>
<th>2.0</th>
<th>PRIMARY COOLANT SYSTEM</th>
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</tr>
</thead>
<tbody>
<tr>
<td>G</td>
<td>Steam Explosion</td>
<td>NC</td>
</tr>
<tr>
<td>H</td>
<td>Natural Circulation</td>
<td>NC</td>
</tr>
<tr>
<td>I</td>
<td>Creep Failure (Steam Generator Tubes, Hot Leg, Surge Line)</td>
<td>NC</td>
</tr>
<tr>
<td>J</td>
<td>Rate of RCS Pressurization During Recovery</td>
<td>NC</td>
</tr>
<tr>
<td>K</td>
<td>Debris Transport to Lower Plenum</td>
<td>NC</td>
</tr>
<tr>
<td>L</td>
<td>External Cooling of RPV Lower Plenum</td>
<td>AE**</td>
</tr>
<tr>
<td>M</td>
<td>RPV Lower Head Integrity</td>
<td>AE</td>
</tr>
<tr>
<td>N</td>
<td>High Pressure Melt Ejection</td>
<td>NC</td>
</tr>
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</table>

* No Change
**Additional Experiments
### 3.0 CONTAINMENT

<table>
<thead>
<tr>
<th>Code</th>
<th>Description</th>
<th>Status</th>
</tr>
</thead>
<tbody>
<tr>
<td>O</td>
<td>Containment Isolation</td>
<td>NC</td>
</tr>
<tr>
<td>P</td>
<td>Containment Bypass (Interfacing Systems LOCA, PWR, SGTR)</td>
<td>NC</td>
</tr>
<tr>
<td>Q</td>
<td>Core-Concrete Thermal Attack</td>
<td>AE</td>
</tr>
<tr>
<td>R</td>
<td>Rapid Steam Formation in the Containment</td>
<td>NC</td>
</tr>
<tr>
<td>S</td>
<td>Overpressure</td>
<td></td>
</tr>
<tr>
<td>S.1</td>
<td>Steam Overpressure</td>
<td>NC</td>
</tr>
<tr>
<td>S.2</td>
<td>Noncondensable Gas Generation from Core Concrete Interaction</td>
<td>NC</td>
</tr>
<tr>
<td>S.3</td>
<td>Burning of Combustible Gases</td>
<td>NC</td>
</tr>
<tr>
<td>S.4</td>
<td>Direct Containment Heating</td>
<td>AE</td>
</tr>
<tr>
<td>T</td>
<td>Overtemperature (Debris-Liner Attack, Containment Penetrations)</td>
<td>NC</td>
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### 4.0 FISSION PRODUCT RELEASE

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<th>Description</th>
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</thead>
<tbody>
<tr>
<td>U</td>
<td>Water Overlying Core Debris</td>
<td>NC</td>
</tr>
<tr>
<td>V</td>
<td>Aerosol Deposition (&quot;Rainout&quot;, Fallout, Containment Sprays)</td>
<td>NC</td>
</tr>
<tr>
<td>W</td>
<td>Fission Product Revaporization</td>
<td>NC</td>
</tr>
<tr>
<td>X</td>
<td>Potential for Suppression Pool Bypass</td>
<td>NC</td>
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</table>

### 5.0 SPECIAL TOPICS

<table>
<thead>
<tr>
<th>Code</th>
<th>Description</th>
<th>Status</th>
</tr>
</thead>
<tbody>
<tr>
<td>Y</td>
<td>Calculation to Estimate the Break Area</td>
<td>NC</td>
</tr>
<tr>
<td>Z</td>
<td>Estimation of Peak Cladding Temperature</td>
<td>NC</td>
</tr>
<tr>
<td>AA</td>
<td>Dual Use of a Single System</td>
<td>NC</td>
</tr>
<tr>
<td>BB</td>
<td>Potential for Criticality of Core Material During Recovery From Severe Accident Conditions (General, BWR, PWR)</td>
<td>NC</td>
</tr>
<tr>
<td>CC</td>
<td>Venting</td>
<td>NC</td>
</tr>
</tbody>
</table>
separate appendix in the TBR and each is organized in the same manner, i.e.:

- Purpose
- Relevant Analytical Information
- Relevant Experimental Information
- Detailed Technical Basis
- Summary
- References

This format was chosen to provide a well characterized and documented technical basis and to provide a convenient means for updating the information should that be necessary at some future date. The information presented in Volume two draws heavily on EPRI-sponsored work in the nuclear industry and universities, NRC-sponsored programs at the National Laboratories and universities, DOE-sponsored efforts at the National Laboratories and universities and work performed under the industry-sponsored IDCOR program. One other major source of information was the Zion probabilistic safety study (CECo, 1981) in which many of the issues related to the details of possible reactor vessel failure and High Pressure Melt Ejection (HPME) were first considered.

As illustrated by the tabulation in Table 2, for most of the issues there is no substantial change from the 1992 status. However, there are substantial changes in several areas as a result of additional experiments. These are:

- External cooling of the RPV lower plenum,
- RPV lower head integrity,
- Core-concrete thermal attack,
- Direct containment heating.

In the next sections we will touch briefly on the information that should be considered as part of the technical basis when developing plant-specific SAMGs.
3.0 UPDATED INFORMATION

3.1 External RPV Cooling

The benefits of external RPV cooling were clearly summarized in the TBR. Moreover, it was discussed that this phenomena could prevent RPV failure under all accident conditions as long as water could be supplied to effectively cool the lower head surface. In this regard, it was noted that plant-specific considerations would include evaluating the potential limitations to water contacting the entire lower head as a result of the vessel support skirt (where appropriate) and the RPV insulation. Experiments at the time indicated that there was no substantive limitation created by reflective insulation and these experiments are still the basis for such evaluations. Experiments performed with a lower head skirt typical of BWR designs in the US (Hammersley, et al., 1993) have clearly demonstrated that the considerations related to the vessel support skirt, and the presence (or absence) of small vent paths to enable steam to escape from the region beneath the skirt, were well founded. In these experiments, the vessel support skirt was observed to develop a stable steam blanket if there was no vent path for steam to escape. When small holes were provided in the simulated skirt, effective cooling of the vessel outer wall was observed. Thus, experimental justification for the TBR considerations has been performed and reported in the open literature.

Other experiments have been performed to examine the heat fluxes that can be removed from curved lower heads under a variety of conditions that could be expected for accident situations (Theofanous, et al, 1994). These experiments demonstrated that the heat removal capabilities from the vessel lower surface are several hundred thousand watts per square meter. Such heat removal rates are well above those in the TBR in terms of the energy fluxes imposed on the RPV wall should molten core debris drain into the lower head. Such external heat removal rates have also been observed in very large-scale experiments sponsored by the Department of Energy and performed at Sandia National Laboratory (Chu, T. Y., et al., 1994a and Chu, T. Y., et al., 1994b). In the Cybl experiments, which approached the size of a full scale reactor vessel and had an elliptical lower head instead of a hemispherical configuration typical of light water reactors, the heat removal rate was again observed to approach 200,000 w/m² which was the maximum energy that could be transferred to the wall in the experiment. These tests observed that the heat removal rate was only by nucleate boiling. Specifically, the small radius of curvature in the reactor system would be expected to provide larger heat removal rates, which is consistent with the observations reported by Theofanous, et al (1994).
In summary, the new information related to external cooling of the RPV further supports the evaluations reported in the TBR. This provides additional confidence for those individuals developing the plant-specific guidelines for severe accident situations.

3.2 **RPV Lower Head Integrity**

There are two major developments in this area since the TBR. The first is the completion of EPRI-sponsored experiments on the integrity of lower head penetrations under severe accident conditions (Hammersley and Henry, 1994) and (Henry, et al., 1994). These experiments tested full-scale, simulated in-core instrument penetrations for both PWR and BWR designs using aluminum oxide to represent the oxidic core debris. In all cases, the molten oxidic material was capable of melting the thimble tube used for the traveling in-core probe (TIP), which exposes a flow path to the containment. However, every experiment demonstrated that the high temperature oxidic material would freeze solid in the process of flowing through the TIP central passage. In fact, these tests demonstrated the central region could reseal completely such that the simulated RPV maintained its pressure for 24 hours. Certainly this observation is consistent with that in the TMI-2 accident where the in-core instrument tubes were damaged, or melted off at the RPV surface and there was no measurable debris transport to the containment (Wolf and Rempe, 1993). Furthermore, large penetrations like the BWR drain line used in some designs were also investigated as possible RPV failure mechanisms. Here again, the experiments demonstrated that the two inch line could be completely filled with high temperature molten material without the structure undergoing substantial strain, i.e., the pipe did not fail. It should be noted that the energy capacity for high temperature alumina oxide is greater than that for uranium dioxide because the specific heat for alumina is approximately twice that of uranium dioxide. Hence, these experiments overestimated the possible energy transfer that could be imposed on the lower plenum structures.

These models have been incorporated into the MAAP4 code (EPRI, 1994) which was designed for supporting accident management evaluations. The TBR presented guidance suggesting that the RPV integrity could be much stronger than had been envisioned in the Zion probabilistic safety study (CECo, 1981) and the IDCOR efforts. These substantial scale experiments, and the experience that none of the penetrations were ejected even though four were completely melted off, provide the necessary insight to support the TBR. Specifically, the current state-of-the-art is that in-core penetrations, BWR vessel drain lines and control rod drive mechanisms would not experience failure shortly after (minutes or tens of minutes) molten core debris would drain into the RPV lower head. This allows substantial time for actions to be taken and the time available for external cooling.
The second issue that has been added to this assessment is the potential for in-vessel cooling by water in the reactor pressure vessel as occurred during the TMI-2 accident. This particular consideration was alluded to in the TBR however there was no substantive information other than the experience that the TMI-2 vessel did not fail. Subsequent to the TBR evaluations, an explanation for the TMI-2 accident behavior has been proposed by Henry and Dube (1984). Two phenomena are particularly important to this cooling mechanism.

- Water availability in the RPV lower plenum when debris would drain into the lower head and the "adherence", or lack thereof, between the molten material and the RPV wall.

- The relative growth of the RPV wall and debris if material creep results from the combination of internal pressure and elevated RPV wall temperatures.

This model has been demonstrated to provide a consistent description of the TMI-2 lower plenum response. Particularly, the wall in the reactor case was observed to increase to a temperature of approximately 1100°C and then cool at a comparatively rapid rate, i.e., about 10°C per minute. The mechanism proposed by Henry and Dube calculates that a limited amount of strain could occur at such elevated wall temperatures and that with a strain of only a few hundred microns, the wall could cool at the rate that was deduced from the post-accident assessment of the TMI-2 RPV material.

This mechanism suggests that once water is returned to the RPV to submerge core debris, the vessel wall would be cooled as a result of limited strain. This is particularly important since this would indicate that water within the reactor vessel is sufficient to stop the accident progression and prevent vessel failure. Hence, for those specific containments where flooding of the containment is either difficult, or of questionable benefit, commitment to such an action would only be necessitated if injection to the reactor coolant system had not been recovered. This particular observation could substantially influence some plant-specific guidances.

3.3 Core-Concrete Thermal Attack

In the technical basis report the potential for cooling of molten core debris in an ex-vessel state was considered in terms of both the WETCOR and MACE tests (Epstein, 1992). These two experiments had substantially different behaviors with the first exhibiting a continuous crust that prevented substantial cooling of the melt, and the MACE test exhibiting an oscillatory type of
behavior with periods of very high heat fluxes (0.6 - 0.8 MW/m²) and other periods of substantially lower heat fluxes (0.1 - 0.2 MW/m²). Additional experiments in the MACE program have been performed but do not enable one to further narrow the possible uncertainties related to cooling in the ex-vessel state. Therefore, this continuing large uncertainty band must be considered by those developing plant specific guidance.

3.4 Direct Containment Heating

At the time that the TBR was published, the Nuclear Regulatory Commission was embarking upon a set of scale experiments to assess the reactor system behavior. The TBR builds a substantial case that direct containment heating in most plants does not provide a credible challenge to containment integrity. The experiments performed by the NRC were done at 1/40-th scale at Argonne National Laboratory (Binder et al., 1994) for a Zion related system and at 1/10-th scale at Sandia National Laboratory (Allen et al., 1994). Furthermore, a similar test matrix was used at both laboratories such that the experiments could be considered counterpart tests, thereby providing a direct demonstration of whether linear scaling is appropriate for this particular phenomenon. These results indeed demonstrated that linear scaling was appropriate and that direct containment heating was not a substitute challenge to containment integrity for the Zion-like containments. Other experiments were performed at 1/10-th and 1/6-th scale, again at SANDIA National Laboratory, for Surry-like geometries (Blanchet, 1994). These tests again showed that direct containment heating was not a credible threat to containment integrity for Surry-like designs and also supports the conclusion that linear scaling was the appropriate means of representing the containment response to this postulated set of severe accident issues.

As part of the analyses for the SANDIA experiments, analytical investigations using RELAP5/SCDAP to calculate the RCS response concluded that extended time to RPV lower head failure would result in hot leg creep rupture well before the vessel would fail. As a result, sequences at the elevated pressure with an uncovered core would not result in high pressure discharge at the time of reactor vessel failure. Thus, from a realistic standpoint, all sequences in which recovery would not occur within the RPV should be judged to be low pressure scenarios for the PWR designs. As a result, the only conditions that could potentially lead to a high pressure melt ejection are those in which injection could be established and the core could be submerged after substantial damage had occurred, like that which was experienced in the TMI-2 accident.

With all the above experimental and analytical investigations, the threat to containment integrity from direct containment heating is reduced from that which was discussed in the TBR. The Technical Basis Report suggests that direct containment heating was not a credible threat to
containment integrity and the continuing work has demonstrated that this conclusion is even stronger than it was in 1992. This should also be understood by individuals formulating plant-specific guidance for severe accident conditions.

4.0 SUMMARY

As discussed in this paper, investigations performed since 1992 support the technical approach presented in the Technical Basis Report. Many of these, such as the potential success of external RPV cooling and the experimental knowledge of DCH related loads, substantially add to the confidence for the technical basis.

In the area of challenges to RPV integrity, significant scale experiments have been sponsored by EPRI to guide severe accident evaluations and the models of accident phenomena in the MAAP4 code. These experiments are a significant addition to the understanding that was documented in the 1992 report. Specifically, these demonstrate experimentally that the penetrations and their support welds are not subject to early failure of the RPV shortly after melt would drain into the lower plenum. Moreover, separate effects and large scale experiments were performed to show the extensive heat removal capabilities from external cooling. Here again it is important that both of these be brought to the attention of those responsible for severe accident management guidance.

In addition, a proposed mechanism for cooling the RPV from the inside by water in the RCS has resulted in a new model for the MAAP4 codes that has been successfully benchmarked with the TMI-2 experience. More importantly, this provides a possible mechanism for stopping the accident progression once ECCS injection has been restored. This possible success path can influence the implementation of the owners group specific severe accident guidelines with respect to flooding the containment.
REFERENCES


Commonwealth Edison Company (CECo), 1981, "Zion Probabilistic Safety Study".


The Status of the Development and Implementation of Severe Accident Management Strategies and Procedures at Ontario Hydro

K.S. Dinnie, G.C. Matthews and J.C. Luxat

Ontario Hydro Nuclear
Toronto, Canada

Abstract

In 1992 a small team was established within Ontario Hydro to establish a technical and conceptual base for preparation and implementation of Severe Accident Management Guidelines. Despite major differences in reactor and containment design with respect to other reactor types, it became apparent that many of the issues to be addressed are similar. Important exceptions are that CANDU severe accidents necessarily progress at low system pressure and, because of the scope of design basis accidents in Canada, existing emergency procedures already deal with the mitigation of some phenomena normally associated with severe accidents, in particular hydrogen production and the need for containment venting. The conceptual base now is largely in place and work to amalgamate the framework with existing symptom-based emergency operating procedures has begun. Preparation and implementation of guidelines has been delayed as a result of a major corporate reorganization and associated resource limitations.
INTRODUCTION

The development of strategies for severe accident management in Canada and in particular at Ontario Hydro, parallel those of other countries in general approach but differ markedly in some of the technical aspects. This arises due both to some fundamental differences in reactor design between CANDU and LWRs and to the fact that some safety issues that are normally thought of as severe accident issues have been addressed as part of the design base in Canada.

In 1992 a small team was established within Ontario Hydro to establish a technical and conceptual base for preparation and implementation of Severe Accident Management Guidelines. This activity is largely complete, although some of the traditionally difficult issues, particularly those associated with steam explosions, cannot be completely resolved.

This paper reviews the issues and general approach to preparation and implementation of severe accident management as they apply to Ontario Hydro. A brief description of those features of the CANDU reactor and the multi-unit containment system relevant to severe accident management is provided, followed by a description of the important characteristics of severe accident progression. The paper focusses on the multi-unit design as employed at its Bruce and Darlington sites.

A comparison is drawn between some of the major accident management issues as they apply to LWR and CANDU systems. Finally, the framework being developed by Ontario Hydro Nuclear to support severe accident management is described with an indication of its current status.

DESIGN FEATURES

2.1 Reactor

The CANDU reactor is a horizontal pressure tube reactor, with a heavy water-cooled, natural uranium fuel contained in some 500 individual fuel channels, passing through a calandria vessel holding the heavy water moderator (Figure 1). The moderator system is low pressure, with bursting discs to form a connection from the calandria vessel to containment in the event of rapid overpressure.

Outside the core, the channels are connected by feeder pipes to common headers, after which the primary heat transport system is fairly conventional in design, except in that some of the major components (e.g. main pumps, steam generators) penetrate the containment envelope for occupational dose control reasons.

Surrounding the calandria vessel is a large shield tank containing light water whose purpose is to provide biological shielding. At each end of the calandria, special end-shields allow access of remote-controlled fuelling machines to the channel end-fittings to enable on-power fuelling. Both the moderator and shield tank/end shield have independent circulating cooling
1. CALANDRIA MAIN SHELL
2. CALANDRIA SUBSHELL
3. CALANDRIA - SIDE TUBESHEET
4. BARRIER PLATE
5. FUELLING MACHINE - SIDE TUBESHEET
6. LATTICE TUBE
7. END FITTINGS
8. FEEDERS
9. CALANDRIA TUBES
10. SHIELD TANK SOLID SHIELDING
11. STEEL BALL SHIELDING (END SHIELD)
12. MANHOLE
13. MODERATOR DISCHARGE PIPES
14. MODERATOR INLETS
15. MODERATOR OUTLETS
16. SHUT-OFF UNIT
17. ADJUSTER UNIT
18. VERTICAL FLUX DETECTOR UNIT
19. CONTROL ABSORBER UNIT
20. LIQUID ZONE CONTROL UNIT
21. END SHIELD COOLING PIPING
22. SHIELD TANK
23. SHIELD TANK EXTENSION
24. RUPTURE DISC ASSEMBLY
25. MODERATOR OVERFLOW
2.2 Containment

A multi-unit station containment consists of four reactor vaults interconnected by a large fuelling duct running immediately beneath the reactors for the length of the station (Figure 2) to permit the access of fuelling machines to the reactors. The fuelling duct serves also as a collection sump for recirculation of cooling water in the event of an accident. The duct is connected to, but normally isolated from, a vacuum building maintained at low pressure whose function is to control containment pressure if isolation occurs due to high pressure or activity in a reactor vault. The vacuum building contains a dousing spray system and can be vented to atmosphere via a pumped, filtered air discharge system.

Each reactor vault is equipped with air coolers and distributed hydrogen ignitors. Most containment systems are initiated automatically following an accident, with the exception of filtered air discharge, which is remote-manual. Air inleakage is expected to deplete the vacuum reserve and lead to a requirement for some limited containment venting in the long term.

3 OVERVIEW OF SEVERE ACCIDENT PROGRESSION IN A CANDU

Progression of accidents in CANDU reactors from those involving little or no fuel damage to significant core damage and possibly core disassembly, is strongly influenced by the unique aspects of the reactor design. In particular, the low pressure heavy water moderator in the calandria vessel surrounding the pressure tubes and the large volume of light water in the shield tank which, in turn, surrounds the calandria vessel, provide a passive heat sink capability which, in many event sequences will provide significant time delays in the progression sequence. Such delays are of benefit in that they provide decision and action time for accident mitigation and management measures to be taken. General characteristics of CANDU severe accident sequences are described below.

A prerequisite for the majority of events to progress to conditions leading to severe core damage is a loss of heat transport system coolant coupled with failure of the Emergency Coolant Injection System, one of the special safety systems, to inject light water coolant into the heat transport system. The loss of coolant may be due either to an initiating pipe break or a consequential induced failure of the heat transport system boundary resulting, for example, from events involving a loss of heat sink.

Other events which can progress to consequential failure of the heat transport system boundary involve impairment of fuel cooling leading to overheating in some fuel channels and resultant thermally induced failure of one channel. It is worth noting that failure of one channel is invariably sufficient to limit further channel failures as a result of the heat transport system depressurization and induced blow down cooling. Furthermore, heat transport system depressurization occurs well before potential formation of molten core conditions, thereby assuring that high pressure melt ejection does not exist as a containment
challenge in CANDU. This conclusion also applies to the very low probability failure to shutdown event [1].

Loss of coolant due to pipe rupture initiating events with coincident loss of the Emergency Coolant Injection System are routinely studied as part of the design basis accident analysis for CANDU reactors. This analysis has demonstrated that, despite severe fuel damage within the pressure tubes, progression of the event to core disassembly is effectively terminated by the passive rejection of heat from the fuel channels to the moderator fluid. This heat sink can be assured for significant periods of time by providing makeup water to the calandria vessel or by establishing coolant injection into the heat transport system.

Should the moderator heat sink be subsequently lost then the moderator will boil off and the fuel channels (pressure tubes and calandria tubes) will heat up, driven by escalating fuel temperatures. Once the temperature of these zirconium alloy components becomes sufficiently high they lose strength and core disassembly will begin. This process, initiated by moderator boil off, will start from the highest elevation channels and gradually progress toward the bottom of the core (Figure 3). However, formation of a core melt pool within the calandria is inhibited by heat rejection from the calandria vessel to the surrounding water in the shield tank (effectively, the calandria is a fully externally flooded vessel).

Failure to add water to the calandria will ultimately result in loss of the shield tank heat sink, either due to boil off of water or a failure of the shield tank. At this juncture, without any mitigating actions to establish some cooling of the core, the accident will progress to core melt formation, melt-through of the calandria and shield tank walls and relocation of molten corium into the containment. However, the containment will retain a large quantity of water consisting of discharged heat transport system coolant and condensed moderator and shield tank liquid. Because of the large surface area for melt relocation and the depth of the pre-existing water pool on the containment floor, effective cooling of relocated melt is expected, with essentially no significant core-concrete interaction.

Event sequence progression in the majority of CANDU severe accidents involves significant periods of time. Progression to core disassembly typically will occur three or more hours after events have progressed to the point that the moderator becomes a heat sink and failure to establish moderator makeup occurs. Further progression to calandria vessel melt through and relocation of melt into the containment will typically occur approximately 24 hours following the initiation of core disassembly [2].

4. SUMMARY OF ACCIDENT MANAGEMENT ISSUES

Table 1 summarizes the major management issues in a severe accident progression arising from sustained loss of heat removal via the primary reactor cooling system/heat transport system (RCS/PHTS), and compares and contrasts the typical LWR and CANDU responses. The presumption is made here that the severe accident results from a prolonged failure of heat removal from the fuel through loss of heatsinks or a LOCA. A third type of CANDU severe accident, although much less likely, can arise from failure of the two independent, fast-acting
Figure 3: Core Heatup and Disassembly

FP release & deposition
suspended debris & fuel
water debris interaction
gas flow pattern
heat loses
terminal debris bed

channel deformation & breakup
shutdown systems to act to prevent an uncontrolled power increase, when called upon to do so. This type of sequence progresses much faster such that core disassembly can occur within a few seconds, causing rapid displacement of moderator coolant. Accident management would be focussed more on maintaining containment functions in such circumstances.

The calandria vessel performs a role analogous to the pressure vessel in severe accident management, in that it can contain the core debris if its integrity can be maintained by internal or external cooling. The significant difference is that this process takes place close to atmospheric pressure in the multi-unit CANDU containment because the calandria is connected to a negative-pressure containment, whereas the pressure vessel remains pressurized to some degree. The result is that the CANDU system is less sensitive to containment bypass and high-pressure melt ejection concerns, because the necessary driving force is absent, at the expense of an early challenge to containment.

The terminal location for the debris if the progression cannot be otherwise halted is the floor of the fuelling duct. The floor will inevitably be covered with water along its length because of the displaced water from the PHTS, moderator and shield tank. Spreading and cooling is expected to minimize corium/concrete interactions (CCI) such that this is unlikely to pose a significant challenge to containment, hence progression is terminated.

5 ACCIDENT MANAGEMENT FRAMEWORK

Emergency Operating Procedures (EOP) were developed to provide procedures to control room operating staff for mitigation of design-basis accidents. New Standards were issued to upgrade the EOPs in 1989, and the upgrade process is nearing completion. Three basic types of accidents are covered by the EOPs: Loss of Regulation (LOR), Loss of Coolant Accident (LOCA) and loss of heat sink accidents. For each of these potential accidents, event-based EOPs are provided for optimal response to the accident.

Symptom-based procedures were introduced in the early 1990s to provide a further layer of defence in mitigating accidents. The symptom-based procedures were to be used whenever the event could not be diagnosed, when multiple equipment and system failures occurred, or when the event-based procedure was unsuccessful. In the symptom-based approach, a small set of critical parameters is identified to monitor and maintain the functions critical to plant safety. Whenever unacceptable values for any of these parameters are encountered during an event indicating a challenge to plant safety, actions are taken to restore the parameters to an acceptable level.
| Decision Point                      | LWR                                      | Multi-unit CANDU                      | Notes                                                                 
<table>
<thead>
<tr>
<th></th>
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<th></th>
</tr>
</thead>
<tbody>
<tr>
<td>Depressurization of RCS</td>
<td>Complex issue with possible negative</td>
<td>Failure of fuel channels occurs early, causing 'automatic' RCS depressurization into calandria vessel and then to containment.</td>
<td>Connection to containment atmosphere minimizes concerns with regard to high-pressure melt ejection, SG tube integrity and containment bypass, but poses challenge to containment.</td>
</tr>
<tr>
<td>Addition of water to degraded core (in-vessel)</td>
<td>Timing and rate of addition determines whether termination or acceleration of accident progression occurs.</td>
<td>Acts to freeze core debris in the calandria vessel but with risk of additional steam surges and hydrogen production.</td>
<td>Maintaining water level in the calandria vessel prior to channel failure assures core structural integrity.</td>
</tr>
<tr>
<td>Addition of water to degraded core (ex-vessel)</td>
<td>Quench debris and scrub fission products, with possible implications to steam generation and hydrogen production.</td>
<td>Fuelling duct floor already covered with water from failed HTS, moderator and shield tank.</td>
<td>Duct can be back-flooded.</td>
</tr>
<tr>
<td>Containment sprays</td>
<td>Cool/depressurize containment and pressure vessel with possible negative implications to hydrogen inverting and long-term containment integrity.</td>
<td>Calandria is fully &quot;externally flooded&quot; by the shield tank. No containment sprays, vent to vacuum building to control pressure.</td>
<td>Shield tank integrity cannot be maintained indefinitely if full decay heat is rejected to it. Dousing spray in vacuum building.</td>
</tr>
<tr>
<td>Hydrogen Igniters</td>
<td>Used in small containments. Concerns from de-inerting.</td>
<td>Automatically initiated to protect against early non-homogeneous distribution.</td>
<td>Steam production aids inverting and promotes redistribution. Covered by existing EOPs.</td>
</tr>
<tr>
<td>Containment venting</td>
<td>Used in some designs to prevent containment overpressure failure.</td>
<td>Pumped, filtered system to control containment pressure when vacuum depleted.</td>
<td>Prevents slow pressurization due to compressed air leakage (or CCI). Covered by existing EOPs.</td>
</tr>
</tbody>
</table>
The international trend in developing Severe Accident Management Guidelines (SAMG) is to extend the symptom based approach to beyond design basis accidents. The symptom-based concept is used to identify challenges to physical barriers and critical functions presented by the accident. Parameters are then identified which either singly, or in combination, are required to monitor those challenges. Strategies to mitigate those challenges would be identified and implemented. An illustration of this method of identifying challenges is shown in Tables 2 and 3. The results of this methodology may then be used in developing symptom-based guidelines.

In the range of severe accidents, one or more of the critical functions would not be controlled by normal means (i.e., the accident is beyond the capability of the EOPs). In such a circumstance, it would no longer be possible to discount feedback effects of the various physical phenomena on the status of safety functions. In addition, the accident itself may render some of the plant indications unavailable. It would be inappropriate, therefore, to address each function on a separate and individual basis as is done in EOPs. Decision making would have to be predicated on the basis of a combination of plant symptoms.

For CANDU, challenges posed by reactor power excursions present a unique circumstance. Reactor power is identified in the EOPs as the critical safety parameter of highest priority. Every available action to restore and maintain this parameter is identified in the EOPs. No further guidelines, therefore, need to be provided as part of severe accident management beyond those described for containment.

Because fuel cooling via the Primary Heat Transport system likely cannot be restored when in the severe accident domain, attention in accident management turns to other objectives. The main objectives of severe accident management are to prevent core dispersal and to prevent containment failure. These objectives are intended to limit dispersal of radioactivity in containment and to limit releases of radioactivity to the public in the event of a severe accident. Tables 2 and 3 are presented to illustrate the use of symptoms to monitor for challenges to plant safety according to these objectives.
### Table 2: Prevent Core Dispersal

<table>
<thead>
<tr>
<th>Challenge</th>
<th>Symptoms</th>
<th>Strategy</th>
</tr>
</thead>
<tbody>
<tr>
<td>Loss of Moderator Heat Sink</td>
<td>Moderator Outlet Temperature, Gross Activity, Containment Pressure, Chemistry</td>
<td>Provide Aux Pump flow, Ensure Service Water supply, Inventory (EWS/ESW), Shield Tank Cooling</td>
</tr>
<tr>
<td>Loss of Moderator Inventory</td>
<td>Moderator Level, Moderator Temperature, Gross Activity, Containment Pressure</td>
<td>Aux Pumps, Inventory (EWS/ESW), Shield Tank Cooling</td>
</tr>
<tr>
<td>Loss of Shield Tank Cooling</td>
<td>Shield Tank Inlet and Outlet Temperature, Gross Activity</td>
<td>Pumps, Service Water Supply, TCVs/Standby HX, Make-up, Vault Atmosphere Cooling</td>
</tr>
</tbody>
</table>

### Table 3: Prevent Containment Failure

<table>
<thead>
<tr>
<th>Challenge</th>
<th>Symptoms</th>
<th>Strategy</th>
</tr>
</thead>
<tbody>
<tr>
<td>Bypass</td>
<td>Ex-Containment Activity (Field Surveys)</td>
<td>Depressurize HTS, Isolate containment</td>
</tr>
<tr>
<td>Core Concrete Interaction</td>
<td>Recovery/Douse Water Chemistry, Containment Pressure, Gross Activity</td>
<td>Flood Reactor Building, Initiate Vault Cooling/EFADS</td>
</tr>
<tr>
<td>Hydrogen Explosion/Deflagration</td>
<td>Containment Pressure, H₂, Igniters Off, PHT Temperature, Moderator Temperature, Air Samples</td>
<td>Maximize Dilution, Initiate EFADS, Igniters</td>
</tr>
<tr>
<td>Temperature Induced Degradation</td>
<td>Vault Coolers Off, Containment Pressure</td>
<td>Restore Vault Coolers</td>
</tr>
<tr>
<td>Energy Addition with Core Failure</td>
<td>Shield Tank Temperature, Containment Pressure, Gross Activity</td>
<td>Prevent Shield Tank Failure/Depressurize, Maintain Moderator/Shield Tank Cooling</td>
</tr>
<tr>
<td>Holes in Containment</td>
<td>Containment Pressure, Ex-Containment Activity</td>
<td>Initiate EFADS, Restore/seal containment integrity</td>
</tr>
</tbody>
</table>
The effective organization and communication between the various groups responsible for responding to a severe accident is considered crucial. These groups will be responsible for using their knowledge base to translate accident management guidelines into an effective procedure for mitigating the accident.

Following an accident, several groups and individuals are required to mitigate the consequences to the public and workers. These groups and individuals follow a formal protocol established for responding to radiation emergencies. Assembled groups are assigned to the Provincial Operations Centre (POC) and the Ontario Hydro Emergency Operations Centre (OHEOC). In addition, municipal authorities, the Local Emergency Management Organization, and the in-plant response groups headed by the Shift Supervisor are also mobilized. These groups are established and coordinated primarily to mitigate the consequences of radioactive emissions from the plant under emergency conditions and to protect the public. Radiation Emergency Procedures are provided to each group to accomplish their respective tasks in protecting the public and workers.

The organizational arrangements for responding to an emergency are already established in Ontario Hydro. In introducing SAMGs, it is important to see that they are implemented to the extent practical using the existing response organization, and without the imposition of the need for undue complexities either in workload of assigned personnel, or in creation of additional groups. This philosophy forms a basis for the production of SAMGs, the role envisaged for the various bodies in implementing them is outlined below.

**Shift Supervisor Responsibilities**

The primary responsibility of the Shift Supervisor following an accident is to implement the EOPs and to coordinate the on-site actions to protect the workers and reduce the potential risk to the public. The Shift Supervisor is provided direction by the Local Emergency Management Organization.

The mandate of the Shift Supervisor is to implement authorized procedures. Any actions which deviate from conventional use of systems and equipment or for which Operating Policies and Procedures may be contravened, would require authorization by the Operations Manager. Implementation of the SAMGs would then be the responsibility of the Shift Supervisor, subject to authorization of the Operations Manager where required.

**Local Emergency Management Organizations (LEMO) Responsibilities**

The Local Emergency Management Organization is located at the station and assembles within 1 to 2 hours of initiation of the accident. The LEMO has functional positions staffed by the following personnel: Station Director, Operations Manager, Engineering Manager, Nuclear Safety Manager, Health Physics staff, radiation control supervisor, emergency preparedness coordinator, community/public relations officer, and local AECB officer.
At Ontario Hydro's nuclear generating stations, different names are applied to this accident response group; the LEMO is referred to as the Site Management Group at Darlington NGS, the Emergency Response Centre at Pickering NGS, and the Station Advisory Group at the Bruce stations. Each group is responsible for coordinating the response effort and managing required resources by:

1. provision of technical and radiological advice
2. directing recovery from the accident
3. communicating and liaising with the corporate emergency centre, off-site centres, and other external organization and the media.

Many of the actions that would be prescribed by the SAMGs would require approval by members of the LEMOs (i.e., Operations Manager, AECB etc) since these actions may require unconventional use of equipment (i.e., use of equipment for other than design intent) and may require mitigating actions to be taken in a range of plant conditions beyond the bounds of the design basis and Operating Policies and Principles. The SAMGs should, therefore, include guidance to the LEMOs in directing the implementation of mitigating actions by the Shift Supervisor to account for contingencies that may arise during the course of an accident.

**Ontario Hydro Emergency Operations Centre (OHEOC)**

This organization is activated within a few hours of the emergency. The primary function of the OHEOC is to support the Provincial Operations Centre with technical information on the accident as needed for off-site decision making in protection of the public. The OHEOC also acts as a technical advisory body to the LEMO. As such, the OHEOC is not expected to play a significant role in implementing SAMGs but may act as a technical advisory body in gathering technical analysis and other support data required. It also has the responsibility for relaying information to the POC and other regulatory authorities.

**Provincial Operations Centre**

This organization is run by Emergency Planning Ontario under the Ministry of the Solicitor General of the Province of Ontario. It is activated within a few hours of the accident. As a provincial organization with jurisdiction for the public's safety, its responsibility is to direct all off-site measures to protect the public, and to provide information to the media. In addition, it may direct the filtered venting of containment in the post-accident phase.

In summary, in the event of a severe accident, the emergency response organization is configured so that the LEMO would oversee and authorize the application of the symptom-based SAMGs by the Shift Supervisor. The LEMO would inform the OHEOC of progress in responding to the accident and the OHEOC would provide technical advice as needed in turn to the LEMO, and would also keep the POC informed of the strategies being implemented.
STATUS OF SEVERE ACCIDENT MANAGEMENT GUIDELINE DEVELOPMENT

A conceptual study for development of SAMGs was initiated in 1992. The key elements of the study included an extension of the symptom-based procedure concept, and the identification of challenges posed by severe accidents. These elements were generally accepted within the corporation as an appropriate approach to be taken for development of SAMGs. A recent corporate reorganization, however, resulted in a slowdown in the work due to associated changes in safety responsibilities and resource limitations.

A draft report outlining a concept for SAMGs was prepared prior to the corporate restructuring. It had limited review and will undergo further review as resources become available. The report recommends a seven step process to be followed in preparing SAMGs. It also recommends that SAMGs be developed to the extent practical on a generic basis. It also develops a framework for extension of the symptom-based concept in the preparation of SAMGs. It is expected that as the EOP upgrades are completed, more effort will be expended in implementation of SAMGs.

CONCLUSIONS

Ontario Hydro is in the early stages of developing and implementing guidelines for the management of severe accidents. A general conceptual and analytical framework has been established and the major decision points identified. Work to amalgamate the framework with existing symptom-based emergency operating procedures has begun but preparation and implementation of guidelines has been delayed as a result of a major corporate reorganization, associated changes in safety responsibilities and resource limitations.

REFERENCES


AMG Products
- General Guidance
- Overview Document
- Technical Support Guidelines (TSG)
- Strategies
- Emergency Procedure Guidelines (EPG)
- Severe Accident Guidelines (SAG)
- EPG Changes
- Implementation Support
- Training Materials
- Training Prioritization Criteria

Overview Document
- TSG
- SAG
- EPG
- PSTG
- Placement Criteria
- Training Program
- Training Prioritization Guidelines
- BWROG products as an integrated set
- Includes 3 products in their entirety
- TSG
- Placement Criteria
- Training Prioritization Criteria
Technical Support Guidelines

- Provide enhancements for optimizing strategies
- Methods should
  - Implement NEI 91-04 guidance
  - Integrate smoothly with E-plans
  - Demonstrate effective decisionmaking, communication, response
- Include 4 interrelated assessments
  - Control parameter
  - Plant status
  - System status
  - EPG/SAG action

Revision 4 EPGs

- Developed into PSTGs and then plant-specific EOPs
- EPG characteristics
  - Symptom-based; event diagnosis not required
  - Avoid precluding future use of equipment if conditions degrade further
  - Use all plant equipment, not just safety-related
  - Attempt to cover all mechanistically possible events
- Approved by USNRC

EPG Changes

- SA information reviewed
- Guiding principles developed to base strategy changes upon
  - Steam explosion
  - High pressure melt ejection/direct containment heating
  - Core-concrete interaction
  - Recriticality
  - In-vessel debris cooling

EPG Changes

- Guiding principles (continued)
  - External vessel cooling
  - Ex-vessel debris cooling
  - Pressure suppression
  - Determination of accident progression
  - Control and termination of accident
Severe Accident Guidelines

- Contains strategies applicable following transition from EOPs to new Integrated Containment Flooding (ICF) strategy
- ICF indicative of onset of core damage (severe accident)
  - Aimed at SA mitigation
  - Uses info not readily available to operators
  - Requires resources outside control room
- Developed into Plant Specific Accident Management Guidelines (PSAMG)

Severe Accident Guidelines

- RPV and Containment Flooding Guideline
  - Uses any one of six injection strategies based on consideration of
    - Water level
    - RPV breach
    - Injection rate
    - Pressure suppression capability
- Containment and Radioactivity Release Control Guideline retains EPG strategies still applicable after transition
- Hydrogen control strategies in SAG

Placement Criteria

- Places EPG/SAG strategies in location where best managed at plant (Control Room or ERO)
- Deterministic, mechanistic, probabilistic criteria
- All must be applied but in any order
- Strategy or specific action not indicated for control room may be directed by ERO
- Placement in ERO not a deviation from EPGs if criteria used

Training

- BWROG developing for utilities
  - Expectations for training program content
  - Proposed training objectives
  - Source documents that can be used
- Generic lesson plans not intended at this time
- INPO training materials will be utilized, but specific relationship with BWROG materials not yet determined
<table>
<thead>
<tr>
<th>Training Prioritization Criteria</th>
</tr>
</thead>
<tbody>
<tr>
<td>Allows plants to prioritize SA training in context with other training needs for both Control Room and ERO personnel.</td>
</tr>
<tr>
<td>Degree of SA strategies in EOPs in equal training based on PSA importance.</td>
</tr>
<tr>
<td>PSAMG trained in initial training or as part of ERO training.</td>
</tr>
<tr>
<td>ERO prioritization criteria based on PSA importance, other factors.</td>
</tr>
<tr>
<td>PSAMG training based on PSA importance, other factors.</td>
</tr>
<tr>
<td>TSG training.</td>
</tr>
</tbody>
</table>
JAPANESE REGULATORY POSITION ON SEVERE ACCIDENT MANAGEMENT AND RELATED RESEARCH ACTIVITIES

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1. INTRODUCTION

Research on severe accident in Japan has been initiated after the TMI-2 accident in 1979. From the regulatory position under Nuclear Safety Commission (NSC), the hydrogen issue, for example, has then been investigated for the possible deflagration/detonation of hydrogen in a containment during severe accident. After the Chernobyl accident in 1986, NSC formed the Adhoc Group on Generic Safety Issues in 1987 to discuss key safety issues related to severe accidents. In 1992, the Adhoc Group submitted to NSC a report on severe accident management.1 NSC endorsed the recommendations by the Adhoc Group and decided to introduce the accident management measures in the statement in 1992.2

In response to this statement, Ministry of International Trade and Industry (MITI) requested utilities to examine possible accident management strategies and to submit reports to MITI.3 MITI strongly encouraged utilities to conduct plant specific level 2 PSA and to initiate the implementation of possible accident management measures based on PSA results. After having conducted systematic level 2 PSAs for all nuclear power plants and investigated technical issues, all ten utilities submitted reports to MITI in March 1994.

After having reviewed those reports, MITI submitted a report to NSC in October 1994.4 In the report MITI stated that the accident management strategies proposed by utilities are generally reasonable, the implementation of the accident management including operational procedures is to be determined and that further research activities are necessary. In response to this, NSC’s review on the report by MITI has been initiated and it is still ongoing.

In accordance with nuclear safety research program by NSC, Japan Atomic Energy Research Institute (JAERI) is conducting a wide range of severe accident research activities both in experiment and analysis. Also Nuclear Power Engineering Corporation (NUPEC) sponsored by MITI is conducting severe accident research focusing on containment integrity related to mitigative accident management and analytical efforts including level 1 and 2 PSAs.

In the present paper, described are the background and status of Japanese regulatory position on severe accident management, and related research activities mostly at JAERI and NUPEC to be utilized for more detailed assessment, implementation, and improvement of the accident management measures.
2. REGULATORY POSITION ON SEVERE ACCIDENT MANAGEMENT

2.1 NSC's Statement on Accident Management

The Adhoc Group established under NSC in July 1987 has been discussing key safety issues, such as the fundamental policy on severe accident, the characteristics of severe accident associated with level-2 PSAs, current design margins, the availability of existing equipment and systems, and possible measures to cope with severe accident. The Group submitted to the NSC "Report on Accident Management as Measures to Cope with Severe Accidents - with Focus on Containment Accident Management" on March 5, 1992. In this report the accident management is regarded as an additional knowledge-based voluntary measures by the utilities to further enhance the safety. The Group also made several recommendations such as NSC to establish a fundamental policy for introducing accident management, utilities to try to implement accident management measures in both preventive and mitigative schemes and to conduct plant specific PSAs, and research organizations and utilities to enhance severe accident research.

NSC endorsed the recommendations by the Adhoc Group and decided to introduce the accident management measures in the statement on May 28, 1992. In the statement, the following instructions are provided:

1. Safety of nuclear reactor facilities in Japan is assured in the existing regulatory practices during each phase of design, construction and operation of nuclear reactor by implementing the rigorous safety measures based on the concept of defense-in-depth, namely, (1) prevention of occurrence of abnormal events, (2) prevention of expansion of abnormal events and their propagation into accident, and (3) prevention of abnormal release of radioactive materials. Consequently possibility of occurrence of severe accidents is extremely small to the level that severe accident would never become reality from the engineering judgment and the risk of nuclear reactor facilities is considered at sufficiently low level.

NSC acknowledges that role of accident management is to further reduce the risk which is already low. Therefore NSC strongly recommends that utilities should prepare effective accident management by their own responsibility and implement them in a timely and appropriate way.

2. Utilities are expected to continuously promote preparation of accident management for further enhancing safety of nuclear reactor facilities by referring to the Subcommittee's proposals.

3. NSC shall review, as necessity arises, reports from the regulatory bodies on measures and implementation plan of accident management. The following practices shall be undertaken by NSC in the meantime.

(1) For nuclear reactor facilities to be constructed in future, the regulatory bodies shall prepare reports on implementation plan of accident management (equipments for accident management, preparation of operation manuals, training of operators, etc.). NSC shall review the report during the process of the safety examination (Double Check) of the nuclear reactor facilities.

(2) For nuclear reactor facilities in operation as well as under construction, the regulatory bodies shall prepare reports on implementation plan of accident management for each plant. NSC shall review the reports.

(3) The regulatory bodies shall prepare reports on the results of PSA of nuclear reactor facilities when the above mentioned (1) and (2) are practiced. NSC shall review the reports.

4. Research Institutions and utilities are encouraged to continue research on severe accident. NSC shall make an effort to acquire and assess results of severe accident research.
2.2 MITI's Requests on Accident Management

In response to the NSC's statement, Ministry of International Trade and Industry (MITI), the competent regulatory body, prepared own policy on implementing accident management to cope with severe accidents and strongly encouraged on July 28, 1992, the utilities to take the following appropriate measures to perform PSA and established PSA-based accident management as an in-house safety assurance measure.

(1) The utilities are requested to conduct level-2 (actually level 2*: level 1 + containment performance assessment) PSAs for internal events during high power operation, clarify the characteristics of each nuclear power plant and examine the possible candidates for accident management measures by the end of 1993. During the examination, the utilities should investigate the technical requirements for accident management at each plant including appropriate operating procedures and training of plant operating personnel.

(2) The utilities are requested to implement accident management measures for each plant promptly and schematically based on the above examination.

(3) Thereafter, the utilities are requested to review of the implemented accident management measures in a periodic safety review.

(4) The utilities are requested to conduct level-1 shutdown PSAs for typical nuclear power plants in Japan within one year and to take appropriate action, depending on the results.

(5) The utilities are requested to further enhance the research on PSA methodology for high accuracy and wider application, and to establish the PSA database, such as on components failure rates.

MITI will require the utilities to report the results of their PSAs as well as the contents of accident management based on them, and will evaluate their validity.

Accident management is a "knowledge-based" action dependent on utilities' technical knowledge aimed at further reduction of the risk which is kept small enough by existing measures, and the utilities are asked to execute accident management in good timing depending on on-going situations by utilizing their best knowledge. MITI, for the time being, have no intention to require any regulatory measures to limit construction or operation of NPPs depending on whether accident management is established at all or on the contents of it.

2.3 MITI's Review on Accident Management

All ten utilities together with vendors have then conducted systematic level 2 PSA for all nuclear power plants (51 NPPs including several NPPs under construction) and investigated technical issues associated severe accident management. All utilities submitted reports to MITI on March 31, 1994 based on those extensive efforts. In these reports, the results of level2 PSA analysis (without source term) have been shown, in which the total core damage frequencies and PCV failure frequencies for all individual NPPs are small enough to meet the IAEA' proposed values for core damage frequency of $10^{-4}$/RY for existing reactors and $10^{-5}$/RY for future reactors and the ones for containment failure frequency which may be practically considered to be $10^{-5}$/RY for existing reactors and $10^{-6}$/RY for future reactors. Also the utilities proposed several accident management measures both in preventive and mitigative schemes based on the PSA results.

MITI has then reviewed those utilities' reports, and MITI submitted a report to NSC on October 24, 1994. In the report the objective of the implementation of accident management measures is stated as to further enhance the safety by appropriately reflecting the recent and
accumulated technical knowledge on PSA and phenomenology on severe accident. In the review process MITI considered the opinions of the expert members in Adhoc Group on Severe Accident Measures established by MITI and the results of PSAs performed by NUPEC were utilized for the comparison. MITI has paid special attention on the suitability of the following three points:

1) Enhancement of safety
   The appropriate accident management measures should be developed against significant accident sequences which are recognized to be taken into consideration by referring to the results of plant-specific PSA(IPE).

2) Feasibility of implementation and effectiveness on prevention and mitigation
   Even though accident management measures are based on the technological knowledge of utilities, they should be by established as engineering measures from the view point of feasibility of implementation, and their effectiveness to the phenomena should be appropriate from the view points of prevention and mitigation.

3) Adverse effects on the existing safety functions
   Accident management measures should be planned not to deteriorate the existing safety functions and not to degrade the prevention level to the design basis events.

As a conclusion MITI stated that the PSA results and the accident management strategies based on PSA are generally reasonable, the implementation of the accident management including operational procedures is to be determined and further research activities are necessary.

2.4 NSC's Review on Accident Management
   The NSC established the Deliberation Committee on Comprehensive Safety of Reactors on September 1, 1994 for the review of issues to further enhance the safety, such as accident management and aging problems. At its first meeting on November 24, 1994 the Deliberation Committee established the Accident Management Subcommittee. The Subcommittee has then started the review of the report by MITI at its first meeting on November 29, 1994. The review process is still in progress and will issue the report to the Deliberation Committee probably by summer of 1995. After having examined the report, NSC will issue the statement with some recommendations, if necessary.

3. RELATED RESEARCH ACTIVITIES

3.1 JAERI's Research on Severe Accident Management
   In accordance with nuclear safety research program authorized by NSC, JAERI is conducting a wide range of severe accident research activities both in experiment and analysis. They cover the fields of the prevention, evaluation and mitigation of severe accidents, such as core degradation behavior, fission product behavior both in coolant system and containment vessel, containment integrity, assessment of accident management measures, and code developmental and assessmental work including PSA. From all these research activities, the following items directly related to accident management are described.

1) ROSA-V Program
   In order to investigate the effectiveness of the preventive measures of the accident management strategies, ROSA-V Program has been conducted at JAERI. The conceptual diagram
of ROSA-V Program is shown in Fig. 1. ROSA-V facility has the capability of full height, full pressure, active steam generators with secondary system, and totally 10 MW of about 1,000 electrically heated rods in core. The feed and bleed measures in the primary and secondary systems are investigated along with the effectiveness of the detectors for the primary thermal-hydraulic conditions. The followings were obtained with a series of experiments; (a) the loop seal clearing occurred during TMLB' sequence involving pump seal leakage, (b) the intentional depressurization of the primary system after the core heatup was found effective to decrease the primary pressure to the accumulator set point, and (c) the effectiveness of the secondary bleed and passive feed procedure, proposed for recovery from TMLB' transient, was demonstrated.

(2) ALPHA Program

For the investigation of the effectiveness of the mitigative accident measures, the molten core coolant interaction experiments have been conducted in ALPHA Program. ALPHA facility and its model containment have the capability of high ambient pressure (2 MPa), high temperature (773 K for electrical cable penetration tests), flexibility of accident management simulation with fairly large amount of melt (maximum 100 kg). In the melt drop steam explosion experiments, the melt was dropped into water and the experimental conditions to suppress the spontaneous steam explosion have been clarified, such as ambient pressure and water temperature.

The conceptual diagram of the molten core coolability experiment is shown in Fig. 2. The melt was generated in the crucible by thermite reaction as in the melt drop steam explosion experiments. The purposes of the molten core coolability experiments are (a) to obtain data on heat transfer from the melt to the overlying water pool to assess the molten core coolability, and (b) to investigate the effect of water injection mode on the interactive behavior between the melt and water. When a stream water was injected onto the melt through a nozzle, eruption of the melt into the water and a violent interaction occurred in one out of seven experiments. However, the melt was cooled without energetic interactions when the water was applied as a spray. It was thus found that the water addition onto the melt as one of accident management measures is generally effective for the coolability of the melt.7

(3) High Temperature Flooding Tests in NSRR

The sequences of severe accident were simulated in NSRR (Nuclear Safety Research Reactor) to investigate in-core fuel degradation due to quenching by water addition in reactor pressure vessel.8 In these in-pile tests, a shortened PWR-type fuel rod was irradiated in NSRR and then heated in a high temperature steam or helium environment. The fuel rod was quenched by water injected into the test section. In a test conducted at high temperature and with high oxidation conditions, the fuel rod was heavily oxidized and was fractured into several pieces by the thermal shock induced by quenching. It was shown that at cladding temperatures above 1773 K, the fuel rod might fail by thermal shock during quenching, even when it had only a thin oxide layer as indicated in Fig. 3. In order to support this experiment, out-of-pile separate-effect tests on core material behavior have also been performed.9 Based on the test results, the reaction rates and the apparent activation energy for each reaction have been obtained. These data are used in computer codes to simulate the fuel behavior during severe accident.

(4) Pool Scrubbing

In a severe accident of an LWR, FPs will be released from the damaged fuel and transported through the primary cooling system to the containment vessel as an aerosol. If, however, a pool of water exists in the release path, such as the suppression pool of a BWR or the pressurizer of a
PWR, FP aerosols would be removed by water scrubbing. The recent PSA indicates that the pool scrubbing efficiency plays an important role in determining the source term of a severe accident with certain kind of accident management measures such as containment venting. Experimental studies for pool scrubbing at JAERI were aiming at pool scrubbing efficiency at elevated pressures and temperatures as experienced at TMI-2 accident using the Experimental Facility for Pool Scrubbing Investigation (EPSI). It was found that the pool scrubbing efficiencies at high pressure (near 7 MPa) and high temperature (near 573 K) is as high as those at low pressure and low temperature.10

(5) WIND Program

The reactor piping would be heated and threatened by superheated steam from the reactor core and by the decay heat of the deposited FPs in severe accident. In order to ensure the accident management measures such as water injections through piping, it is important to evaluate the integrity and to demonstrate the safety margin of piping. To meet these objectives, FP aerosol behavior tests and high temperature piping behavior tests are being performed in the WIND(Wide range piping INtegrity Demonstration) Program.11 The FP aerosol behavior tests will introduce nonradioactive CsI and CsOH, and the behavior of aerosols, such as deposition, revaporation and resuspension, will be investigated.12 The schematic diagram of aerosol behavior test is shown in Fig. 4. The high temperature piping behavior tests will demonstrate the integrity and the safety margin before the failure of various types of reactor piping. The first phase of both tests will be initiated in FY 1995.

As a preliminary work, the WAVE(Wide range Aerosol VERification) experiments have been conducted to investigate the CsI aerosol behavior in a piping. A stainless steel test tube having an inner diameter of 40 mm and a length of 1000 mm was used in the tests. It was found that the coupling of FP behavior and the detailed thermal-hydraulic analysis is essential to accurately predict CsI deposition, primarily due to three-dimensional flow in the piping.13

(6) Analytical Work

The code developmental and assessmental work is conducted for an integrated source term analysis code, THALES-2, which has been used for level-2 PSA. Also the fission product analysis code, ART, and containment aerosol analysis code, REMOVAL have been developed at JAERI. Application of detailed and mechanistic codes mostly introduced from USNRC, such as SCDAP/RELAP5, CONTAIN, and CORCON, to experimental analyses has made progress by participating in international standard problems and code comparison exercises.

An accident management analysis during a station blackout sequence with a PWR pump seal LOCA (S3-TMLB') was performed with SCDAP/RELAP code. The results showed that the accumulator injection does not initiate automatically during S3-TMLB' and that the intentional depressurization using PORVs should be recommended as a countermeasure to mitigate the consequences of the accident.14

The effectiveness of the reflooding action of the damaged core by the operators has been parametrically evaluated with SCDAP/RELAP code for the simulated TMI-2 situations. It was found that the fuel cladding temperature excursion is obtained due to increased metal-water reaction, and that the reflooding action should be done as soon and as much as possible.15

3.2 NUPEC's Activities on Severe Accident Management

NUPEC is conducting the Containment Integrity Test to reduce the unknowns regrading the
mitigative accident management such as evaluation of peak pressure during hydrogen burning, ultimate strength and leakage conditions in the containment vessel, effectiveness of fission products removal under accident management conditions and fuel-coolant interaction during debris cooling. NUPEC is also conducting a program for developing methodology of level-1 and 2 PSA and severe accident analysis. From all these activities, the following items directly related to accident management are described.

(1) Hydrogen Mixing and Combustion Tests

The hydrogen behavior tests, namely hydrogen mixing and distribution test, and hydrogen combustion test, have been conducted in order to understand fundamental phenomena of hydrogen mixing and combustion in containment vessel and to evaluate containment integrity against hydrogen combustion under severe accident conditions. The hydrogen mixing and distribution test facility is 1/4 scaled Japanese 4-loop PWR model with 1,600 m³ model containment vessel in volume with 25 compartments as illustrated in Fig. 5; Helium gas in stead of hydrogen was used in the tests. It was found from the tests that the hydrogen concentration in a containment dome is expected to be almost uniform for Japanese PWRs mostly due to loose connection between compartments.

The small scale and large scale combustion (deflagration) tests have been conducted at NUPEC. The test facilities for these tests are shown in Fig. 6. In the small scale test with 5 m³ test vessel, the design pressure and temperature are 3 MPa and 573 K, respectively. In the large scale test with 270 m³ test vessel, the maximum hydrogen concentration is 18 %. It was found from the combustion tests that the maximum pressure in the containment dome is expected to be about 0.5 MPa under hydrogen combustion without steam and spray for Japanese PWR dry containment.

(2) Structural Behavior Tests

Tests to failure of scaled steel and concrete containment vessel models are planned under a joint research program involving the NUPEC, the USNRC and the Sandia National Laboratories (SNL). The objectives of these tests are to measure the failure pressure, to observe the mode of the failure, and to investigate the ultimate structural behavior of the containment vessel models by slowly increasing the internal pressure, at ambient temperature, until failure occurs. Also, pre- and post-test analysis will be performed to predict and evaluate the test results, and to validate the analytical methods that will be used to evaluate the structural behavior of the actual containment vessels under severe accident conditions. The steel containment vessel (SCV) model was fabricated in 1994 and was transported to SNL for instrumentation in March, 1995. The over-pressure test of the SCV model will be conducted in fall, 1996. The fabrication of a pre-stressed concrete containment vessel (PCCV) model will be started in 1995. And the test of the PCCV model will be conducted in 1999.

(3) Fission Products Removal Tests

Organic resins or silicon rubber have been used as seal materials in the penetrations of electrical cable assembly and flange/gasket of the hatch in the containment. These penetrations have a possibility to produce leakage path even below the sustaining pressure limit due to aging by over heating/or radiation exposure under severe accident. In order to evaluate the failure criteria of the seal materials in the containment penetrations by over heating, mock-up of real electrical cable assembly and hatch are prepared to study leak characteristics by heating under high pressure. Further, aerosol materials would be trapped in leak path and the possibility to reduce source term to the environment.
Before the study by mock-up apparatus, fission product trapping effect in leakage path in small scale has been studied in the various parameters and evaluated aerosol trapping mechanism using a capillary or a flange/gasket test section geometry. Gaseous iodine (I₂, CH₃I), soluble aerosol (CsI) and insoluble aerosol(Fe₂O₃) are introduced into the leak path under severe accident conditions (< 400 °C, <10⁴ Pa and < 80 vol.% steam fraction) to study fission product penetration and trapping behavior in it.

As a summary of small scale test, plugging occurred within 1 to 3 hours near inlet region of leak path. Plugging time became shorter with increasing aerosol concentration and depended on capillary inlet geometry.

In the containment accident management measures, alternative injection of water by containment spray is being planned to cool the atmosphere. Especially in BWR, due to the relatively small containment volume, containment venting will be conducted in ultimate situation. Containment spray also provide an effective measure to remove fission product in atmosphere except inlet gas.

In order to provide experimental data for demonstrating the effective aerosol FP removal by containment spray under accident management conditions and for the assessment of analytical codes, FP removal tests have been conducted. The test facility, originally GIRAFFE used for passive cooling system demonstration of SBWR, has a full height and 1/400 scaled volume of BWR for the integral simulation of both thermal hydraulic transient and FP transport behaviors. The test conditions will be determined mostly based on the level-2 PSA results.

The thermal-hydraulic condition of containment during the accident management procedure are under low flow spray, high humidity due to debris cooling by water and long term fresh water supply. Aerosol particles as cesium iodide or cesium hydroxide have grown its particle size by absorbing steam even in unsaturated atmosphere due to their hygroscopic characteristics. In this atmospheric conditions, aerosol settling would be further accelerated due to particle growth. The test includes separate effect tests, such as humidity effect test to provide hygroscopic aerosol removal characteristics under high humidity conditions, and natural removal test to investigate natural removal of FP aerosols by gravitational settling and diffusion to the wall.

(4)Debris Cooling Tests

In order to confirm the coolability of debris both retained in the pressure vessel and ejected onto the lower part of the containment vessel, NUPEC participates in the RASPLAV Project organized by OECD/NEA, and also conducts the COTELS program.

Through the RASPRAV project, NUPEC intends to obtain the thermal properties of the molten corium retained on the lower head, the thermal interactions between the molten corium and the lower head, and the material properties composed of uranium, zirconium, stainless steel and their oxides.

NUPEC is conducting the COTELS project in order to evaluate the coolability of debris ejected from the lower plenum onto the lower part of the containment vessel. Considering the accident sequence evaluated by accident analysis codes, for instance, MAAP, the COTELS project consists of main three tests, i.e., the test in order to investigate the interactions between the molten debris and water accumulated in the lower part of the containment vessel, the tests in order to evaluate the coolability of debris by the water injection, and the tests in order to understand interactions between debris spread on the lower part of the containment vessel and the concrete, under the accident management condition that water is injected onto the debris.
(5) PSA Study and Severe Accident Analysis

Institute of Nuclear Safety (INS) of NUPEC has been conducting level-2 PSA studies and severe accident analysis, under the sponsorship of MITI, which are used as reference PSAs in the review on IPEs by utilities, to supply probabilistic safety information related to severe accidents to the competent regulatory authorities. In this program, MELCOR code is extensively applied to analyze accident progressions of dominant sequences, and further to examine the effectiveness of accident management measures. Applications of MELCOR code to PSA have been conducted for both 1,100 MWe BWR with Mark-II containment and 1,100 MWe PWR with large dry containment design. Although MELCOR is a versatile tool to reasonably calculate the accident sequences, it was found that further validation and improvement of the models are necessary.

The effectiveness of automatic depressurization system (ADS) as a preventive accident management for BWR-5 with Mark-II containment has been evaluated with MELCOR. The TQUX sequence, a transient with loss of high pressure core spray systems, was identified one of the dominant sequence that lead to core melt, and parametric study of the ADS activation time has been conducted (Fig. 7). It was found from the analysis that the suitable ranges of the depressurization timing of the reactor coolant system exist to prevent core damage.

The effectiveness of water injection into containment as a mitigative accident management for four-loop type PWR with large dry containment has also been evaluated with MELCOR. The ADC sequence, a large break LOCA with the loss of high and low pressure injection, and containment spray systems was identified as one of the dominant sequence that lead to containment failure, and parametric study of heat transfer model has been conducted. It was found from the analysis that the water injection into containment is generally effective to mitigate the accident sequences (Fig. 8), but there still exist large uncertainties involved in the cavity concrete ablation.

3.3 Industries' Activities on Severe Accident Management

The industries have also made extensive efforts on the evaluation of accident management measures, as well as the systematic level-2 PSA analysis for each plant, under the scheme of cooperative study among electric utilities with vendors. Those include experiments on the core melt spreading on the containment floor, and on the pool scrubbing efficiency in a BWR pressure suppression pool.

In developing the accident management strategies, ex-vessel core melt coolability is one of important issues. In the SPREAD experiment, molten stainless steel as molten core simulant was spread on the concrete floor which simulates pedestal region with and without water pre-existence conditions. The systematic diagram of the test section is shown in Fig. 9. It was found from the experiments that the melt spreading was largely reduced by the presence of water and thus water supply on concrete floor was thought to be an effective accident management measure.

Aerosol removal by pool scrubbing in the pressure suppression pool is a characteristic safety feature of BWR. Although the probability is quite low, this effect is expected even when the core is severely damaged and containment venting is actuated. The pool scrubbing experiment in the industry utilized the test section of 1 m in diameter and 5 m in height as shown in Fig. 10. It was found that the pool scrubbing efficiency is much affected by the pool depth, aerosol diameter, and steam content. The correlation of pool scrubbing efficiency has been developed based on this experiment.
4. USE OF RELATED RESEARCH ACTIVITIES IN REVIEW PROCESS

The results of the related research activities such as JAERI, NUPEC and industries have been effectively utilized in the review processes of accident management at MITI and NSC. In the report by MITI to NSC described in 2.3, the effectiveness of the proposed accident management measures has been evaluated based on the state-of-the-art technical knowledge obtained worldwide. Especially for the following three items, the results obtained in Japanese activities have been referred in the report as follows:

(1) The steam explosion is the phenomena in which the thermal energy of high temperature material is transferred immediately to the mechanical energy, when the high temperature melt is fragmented during the process of interaction with water. It has been clarified with JAERI and other experiments that certain conditions, such as high ambient pressure or high water temperature, can effectively suppress the occurrence of the steam explosion.

(2) The hydrogen gas are mostly generated from the fuel clad-water reaction and water decomposition by radiation. The combustion of this flammable gas at high concentration may lead to the high pressure loads to containment vessel. For the ice-condenser containment type Japanese PWR igniters are to be implemented to mitigate this effect. According to NUPEC and other tests, it was found that the mixing in the containment is well and that the energy released at low concentration is relatively small.

(3) The melt attack is the phenomena in which the melt spreads to the containment side wall and it ablare the side wall by its high temperature. According to the experiment in the utilities' cooperative research, it was found that this phenomena can be prevented by effectively removing the decay heat by the existence of water.

The MITI's report also emphasizes the importance to further clarify the occurrence condition and consequences of these phenomena, and to improve the analytical models in the codes for high accuracy in order to implement more effective accident management measures.

The above described activities including analytical efforts are also being utilized in the review process of MITI's report by NSC. It is also noted that the future activities to further encourage and assure the implementation of accident management measures will be recommended.

5. SUMMARY

In response to the statement on severe accident management by Nuclear Safety Commission in 1992, MITI requested the utilities to examine possible accident management strategies. After having conducted systematic level 2 PSA for all nuclear power plants and investigated technical issues, the utilities submitted reports to MITI. MITI reviewed the technical appropriateness of them and then submitted a report to NSC in 1994. Therefore the review process by NSC is now ongoing. It has been clarified again through these review process that the severe accident management is not a licensing requirement from a regulatory position, but is the additional measures based on the generalized technical knowledge with maximum flexibility voluntarily implemented by utilities to further enhance the safety.

In the review process by MITI and NSC, the related research activities at JAERI, NUPEC, and industries have been utilized to evaluate the effectiveness of the proposed accident management
measures along with the findings from foreign activities. Since there still exist relatively large uncertainties involved in some severe accident phenomena, such as steam explosion or melt coolability, it should be noted that further research is necessary to support and improve the effective accident management implementations. Also these findings from severe accident research will greatly contribute to the development of next generation nuclear reactors.

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Fig. 1 Conceptual diagram of ROSA-V

Fig. 2 Conceptual diagram of melt coolability experiments in ALPHA

Fig. 3 Failure map of high temperature flooding tests obtained at NSRR

Fig. 4 Aerosol behavior test facility in WIND
Fig. 5 Hydrogen mixing and distribution test facility

Fig. 6 Small scale and large scale hydrogen combustion test facilities

Fig. 7 Effect of depressurization timing on fuel temperature for TQUX sequence

Fig. 8 Effects of water injection into CV on accident progression for ADC sequence
Fig. 9 Schematic diagram of SPREAD test section

Fig. 10 Schematic diagram of pool scrubbing test
SEVERE ACCIDENT MANAGEMENT

PRESENTATION OF
AND
TECHNICAL BASIS FOR
THE
CURRENT EDF APPROACH

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1)-Introduction

French current plants have not been designed to explicitly address Severe Accident issues.

Originally, starting from the identification of key safety functions to be fulfilled, i.e.:

- reactivity control
- decay heat removal from the core and to the environment
- prevention of fission product release to the environment

safety systems were designed to allow compliance with adequate acceptance criteria in all normal or perturbed situations identified in the Design Basis.

As was the case in most foreign countries, this approach had two very important characteristics:

- only initiating events with single malfunctions or failures were considered,
- deterministic plant safety assessment was made under the key assumption that essential safety grade systems had their minimum required operating capability.

From a very formal standpoint, this meant that situations leading to core melt were not explicitly contemplated, even though the containment system had the capability to provide adequate protection for many core melt sequences.

At the accident management level, Emergency Operating Procedures (EOPs) were provided to deal with incidents and accidents and consequently prevent core melt situations (1 and A procedures in French). All actions required from the operators for proper handling of the situation being based on systems operating as designed, no critical action was needed to address all situations bounded by DBAs.

Very soon however, it became clear that more complex situations dealing with multiple failures, loss of redundant systems, or complete loss of functions having the potential for leading to core melt and (or) containment failure could also exist.

Depending on their nature, associated challenges were addressed considering the capability of existing systems, the intrinsic resistance capability of plants as designed and reasonable modifications.

The key objective being to prevent core melt, plant response was improved in case of degraded situations through:

- use of already existing systems, extending their domain of operation to allow recovery of essential safety functions,
- plant modifications allowing to better cope with well identified challenges and provide time to recover one essential safety function (e.g. dedicated steam-driven turbine for water supply to RCP seals in case of Station Blackout)

The most important challenges considered were:
- Anticipated Transients Without Scram (ATWS)
- Station Blackout (SBO)
- Total Loss of Heat Sink
- Total Loss of Feedwater flow (Main and Emergency Feedwater)
-Loss of the Low Head portion of the Safety Injection System or of the Containment Spray System

At the accident management level, additional EOPs were implemented to allow the operator to deal with these situations. Though being an extension of preexisting EOPs, these new procedures addressing Beyond Design Basis situations (H procedures in French) differed slightly from I or A procedures in that some critical actions such as depressurization of the Reactor Coolant System (RCS) were sometimes required to reach a state where all safety functions can be fulfilled.

At last, for even more complex challenges where one safety function could be entirely lost, other procedures (U in French) were also implemented and guidance provided either to try to prevent further evolution of the situation towards coremelt or mitigate the consequences of core degradation through guaranteeing no uncontrolled loss of containment integrity. These procedures or guidance also required critical decisions to be made in order to limit the consequences of the accident outside the plant.

N.B. Originally, all EOPs (I,A,H) were Event Oriented. The U1 procedure was the first Physical State Oriented Procedure developed to address challenges to the core and the RCS, other U procedures (U2 to U5) addressing Station Blackout or potential loss of containment integrity. To have a more coherent approach, EDF decided to implement a complete set of State Oriented procedures to replace EOPs and U1 for the last 4-loop plants, and to extend the approach to all other 3 and 4-loop plants later.

As differences between both approaches in making critical decisions are minor, explanations will be given for Event Oriented procedures only as implemented for 1300MWe plants. Also, for clarity, only at power situations will be dealt with though the approach in fact covers shutdown states.

II) General Organisation for Accident Management

For Accident Management, all actions are performed by the operator according to what is required by EOPs, State Oriented Procedures or Severe Accident Guidelines. However, it was felt necessary to provide some redundancy to operator diagnosis to limit the risk of improper action. Redundancy is provided through monitoring of the status of key safety functions: this monitoring is performed by an Engineer identified below as the Safety Engineer (ISR in French).

Figure 1 shows the general organization in case of accident and the respective roles of the operator (actions) and the Safety Engineer (monitoring). This sketch also clearly shows that the Safety Engineer can interrupt the implementation of EOPs after analysis of the evolution of the accident and require implementation of U procedures.

Also, though not mentioned in this sketch, he can recommend to the local crisis center termination of U procedures (essentially U1) and shifting to implementation of Severe Accident Guidelines when the situation is degrading further.

III) Monitoring key safety functions

3.1) In case of abnormal operation, the Safety Engineer has to monitor key safety functions. This is done through the implementation of a Permanent Monitoring after an Incident (SPI in French), or during degraded situations (SPU in French)
Considering the design of current French plants, the most important systems allowing to fulfill these key functions are:

- core criticality:
  * control and shutdown rods
  * Safety Injection System for Boron Injection in case of accidents

- Decay Heat Removal from the core:
  * Safety Injection System for LOCA and some non LOCA events
  * Reactor Coolant Pumps

- Decay Heat Removal to the Environment
  * Steam Generators and Emergency Feedwater System

- Confinement of fission products
  * Containment System

As indication of operation of systems and components as expected is made through informations from the I&C system, a monitoring of the status of the system supply is also performed to identify possible failures and make key assumptions in case of malfunctions of subsystems needed for operation of non redundant sensors.

Figures 2 (for SPI) and 3 (for SPU) describe briefly the organization of this monitoring.

In subparagraphs below, only the portion of monitoring which is meaningful for making critical decisions will be dealt with.

### 3.2) I&C supply monitoring

The most important malfunctions the Safety Engineer has to deal with are that leading to the loss of non redundant informations. Dealing with operation at power, these informations are:

- Activity in the Steam Generators
  In case this information is lost, the operator assumes the activity value is that prevailing before I&C malfunction.

- Steam Generator level (wide range)
  These informations can be recovered through realignment of another system. Otherwise, partial redundancy is provided through the informations from narrow range level channels.
  If wide range information is unavailable and the level is outside the narrow range span, the Steam Generator is considered empty.

- Control Rod drop indicators
  If not available, it is assumed that at least two control rods are stuck in the fully withdrawn position.

- Neutron Flux (Intermediate Range)
  Assumed unavailable in case of supply malfunction.

All other informations having redundant supply, the Safety Engineer will request corrective action on the affected part of the supply system in case of partial loss of information.
These malfunctions are of interest only because they can influence the Safety Engineer diagnosis concerning system or component availability.

3.3) Core criticality monitoring

Normally, criticality control is made through rod insertion in the core and boron injection.

The only malfunction leading to critical decisions is an ATWS. In this case, more than one rod has not been inserted in the core.

The first action requested is the interruption of power supply to the control rods to allow rod drop in the core. Then boration of the RCS is requested, if boration systems are available.

3.4) Monitoring of Decay Heat Removal to the Environment

For non LOCA events, the frontline systems for Decay Heat Removal to the Environment are the Steam Generator System and Feedwater Systems (mainly Emergency Feedwater). The only question of interest for making critical decisions is the availability of the Steam Generators.

Steam Generators can be unavailable in two cases:

*the activity content of the secondary side is too high
*the secondary water inventory is too low and no recovery is deemed possible.

3.4.1 The activity content can rise in the Steam Generator after a loss of integrity of one (or more) SG tube.

Evaluations made in EDF have shown that fourth category acceptance criteria were not exceeded when the Iodine Equivalent Concentration in the secondary system was below a few Curies per ton of water, e.g. for a 4-loop plant:

*260 Cl \(_{131}\)/ton when the SG level is controlled in the narrow range
*15 Cl \(_{131}\)/ton when the SG is not controlled, i.e. the level is above the upper limit of the narrow range.

If \(C_0\) is the activity concentration in the SG prior to the accident, the SG is considered:

*available if the activity in the secondary water is below 100 \(C_0\)
*not available but reusable if needed if the activity is in between this value and the maximum activity which can be measured by the Sampling Systems connected to the SG Blowdown System (\(10^{-1}\) Cl/ton)
*unavailable in all other cases (not reusable)

Figure 4 summarizes these considerations

3.4.2 Concerning water inventory, indications allowing to make important decisions are the Steam Generator Levels. In practice, attention is paid to the lower limits of both the narrow and the wide range, accounting for measurements uncertainties.

As an exemple, for 4-loop plants:

*if the level is above 5% (narrow range), or if the level is below 5% (narrow range) and above 17% (wide range), the SG is not considered empty and unavailable.
*if the level is below 17% (wide range), the SG is considered empty.

3.5)-Monitoring of Decay Heat Removal from the Core

Degradation of the conditions in the RCS is detected through monitoring of the following parameters:

- margin to saturation, defined as $\Delta T_{\text{sat}} = T_{\text{sat}} - T_{\text{RCS}}$ (where $T_{\text{RCS}}$ being the temperature at core outlet).

Important values of this parameter are:

* $10^\circ$C: below this value, the RCS is considered at least saturated, meaning that the water inventory in the RCS is degraded
* $-30^\circ$C (at least one Reactor Coolant Pump (RCP) operating). Above this value and if there is enough water in the vessel ($\Delta P$) the RCS could be saturated. On the contrary, below this value or if there is not enough water in the RCS ($\Delta P$), the RCS is considered supersaturated.

- temperature at core outlet.

* curve in figure 5 (all RCPs stopped): this curve is an envelope of the evolution of the core outlet temperature as a function of time for all accident sequences considered in the design or for Beyond Design sequences identified as H.
* $320^\circ$C: saturation temperature corresponding to the opening setpoint of the SG safety valves
* $355^\circ$C: saturation temperature corresponding to the opening setpoint of the last RCS safety valve.
* $700^\circ$C: the RCS inventory is supersaturated, meaning that the situation is significantly degraded, but there is still significant margin before the onset of core melt.
* $1100^\circ$C: this is an envelope of the temperature in case of a controlled Design Basis Large Break LOCA.

N.B. Availability of the Safety Injection System is a key element for making critical decision and is monitored through analysis of level information in the Refueling Water Storage Tank and in the containment sumps

3.6)-Containment System Integrity monitoring

The objective during an accident being to prevent dispersion of fission products in the environment, containment integrity is monitored using pressure and activity measurements in different areas of the plant where fission product release could occur.

Dealing with activity, essential parameters and values are (for 4-loop plants):

- activity measured in the stack of the auxiliary building
  * greater than $3.7 \times 10^6$ Bq/m$^3$ during at least 30 minutes or
  * instantaneous value greater than $2.10^9$ Bq/m$^3$

These values characterize an accident not enveloped by the most limiting DBA with the maximum containment leak rate.

- activity of the RCS water greater than 0.5 rad/hr.
The activity of the RCS water is greater than that evaluated for the most severe DBA
-activity measured in the sumps of either the fuel building or the nuclear auxiliary
building greater than 20 rad/hr.
This characterizes a leak on a safeguard systems and the value is exceeding the
maximum corresponding to the capability of the Waste systems.

-γ activity in the containment atmosphere above 10^{4} \text{ rad/hr}.
This value is above what is anticipated for the most severe DBA, assuming 100% cladding
failure and unavailability of the Containment Spray System.

IV-CRITICAL ACTIONS

4.1-Critical actions in EOPs

In some Beyond Design situations, critical actions are sometimes required, dealing mainly
with Total Loss of Feedwater and ATWS:

4.1.1-Depressurization of the RCS
This is required when Steam Generators are unavailable due to a complete loss of
Feedwater Flow and the RCS is still undersaturated (see § 3.5 for adequate criteria) and
the Safety Injection System is available. RCS depressurization so allows to decrease the
pressure below the shutoff head of the Medium Head Safety Injection system and recover
the capability to inject water into the RCS and cool the core.

4.1.2-Depressurize Steam Generators
Decay Heat Removal to the environment is normally made through steam release on the
secondary side of the Steam Generators.
However in some situations, the Steam Generator is empty while water is still supplied.
SG pressure is then tested: if it is below the opening setpoint of the atmospheric dump,
the SG is isolated by the operator, and inspections are made to detect a leak outside
containment, if any. If no leak has been found, the Safety Engineer requests
investigations to try to realign other circuits for SG supply resumption.
If the pressure is above 80 bars, and there is a potential for injecting excess reactivity
in the core (which could be the case for an ATWS), the depressurization capability is
limited to that needed for decay heat removal only, and the SG is temporarily considered
unavailable.

4.2-Shifting from EOPs to U procedures

Through monitoring of the situation in the RCS and other important systems or buildings, the
Safety Engineer has the possibility to detect a degradation of plant essential safety functions.

In this case, he can request the operator to implement:

-the U1 procedures, in any one of the following cases:

*no SG available, and at least one with "high" activity
*no SG available and Pressurizer relief valves closed and Temperature at core outlet
above 320°C
*RCS fluid at saturation or supersaturated (see § 3.5) and:
Safety Injection not available

-the U2 procedure (related to containment system behavior), after reaching one of the criteria mentioned in § 3.6

.RCPs stopped and T at core outlet above the envelope curve (see Fig. 5)
.at least one RCP operating and RCS fluid supersaturated or water inventory in the RCS low.
*low water inventory in the Refueling Water Storage Tank and the Containment sump.
*activity inside containment high

4.3-From U procedures to Severe Accident Guidelines

When the situation is degrading even further and it seems difficult to prevent the onset of coremelt without further action. Prevention of containment catastrophic failure and limiting external releases the pressure in the RCS is decreased either to try and inject water in the RCS in case of injection system recovery or go to low pressure coremelt sequences to limit risks of RCS failure or Steam Generator Tube Ruptures. Ultimately, implementation of the U5 procedure can be recommended. Criteria to be met for implementation of Severe Accident Guidelines or the U5 procedure are:

4.3.1-Severe Accident Guidelines

*T at core outlet above 1100°C
*activity inside containment above that mentioned in Table 1

<table>
<thead>
<tr>
<th>Time after Scram</th>
<th>Setpoint</th>
</tr>
</thead>
<tbody>
<tr>
<td>t &lt; 1 hr</td>
<td>s = 500 Gy/hr</td>
</tr>
<tr>
<td>1 hr &lt; t &lt; 6 hrs</td>
<td>s = 100 Gy/hr</td>
</tr>
<tr>
<td>6 hrs &lt; t &lt; 5 days</td>
<td>s = 50 Gy/hr</td>
</tr>
<tr>
<td>5 days &lt; t &lt; 1 month</td>
<td>s = 10 Gy/hr</td>
</tr>
<tr>
<td>t &gt; 1 month</td>
<td>s = 5 Gy/hr</td>
</tr>
</tbody>
</table>

These values correspond to the maximum activity inside containment evaluated for the most limiting LOCA, with 100% failed fuel cladding and no Spray System actuation.

Upon reaching one of these criteria, plant headquarters can decide to leave EOPs and implement Severe Accident Guidelines.

4.3.2-U5 Procedure

*implementation of Severe Accident Guidelines and
*at least 24 hours into the accident and
*low pressure buildup inside containment and
*containment pressure between 5 and 5.5 bars (preferred values) in no case greater than 7 bars (characteristics of French concrete containments).
4.4 Depressurization of and Water Injection into the RCS

Though water injection into the RCS is standard procedure in case of DBAs, decision to inject water could be critical for more perturbed situations. Also, the capability to inject water into the RCS can have a major influence on the decision to depressurize the RCS, action which is considered critical in current plants.

Table 2 summarizes criteria and conditions for making these decisions

<table>
<thead>
<tr>
<th>Parameters</th>
<th>At least one SG available</th>
<th>No SG available or reusable</th>
</tr>
</thead>
<tbody>
<tr>
<td>$\Delta T_{\text{Sat}} &gt; 10^\circ\text{C}$ and $T_{\text{core outlet}} &lt; 355^\circ\text{C}$</td>
<td>Inject Water(SIS)</td>
<td>Inject Water(SIS) Depressurize if SIS available</td>
</tr>
<tr>
<td>$-30^\circ\text{C} &lt; \Delta T_{\text{Sat}} &lt; 10^\circ\text{C}$ and $T_{\text{core outlet}} &lt; 355^\circ\text{C}$</td>
<td>Inject Water(SIS)</td>
<td>Inject Water(SIS) Depressurize if SIS available</td>
</tr>
<tr>
<td>$\Delta T_{\text{Sat}} = -30^\circ\text{C}$ or $T_{\text{core outlet}} &gt; 355^\circ\text{C}$</td>
<td>Inject Water(SIS) Depressurize if $T_{\text{core outlet}} &gt; 700^\circ\text{C}$</td>
<td>Inject Water(SIS) Depressurize</td>
</tr>
</tbody>
</table>

$\Delta T_{\text{Sat}} = T_{\text{Saturation}} - T_{\text{core outlet}}$

Discussion

-depressurization

*decision

The RCS is depressurized when:

1-Steam Generators are not available or reusable and the RCS is at most slightly supersaturated (see § 3.5 for adequate criteria) and the Safety Injection System is available
2-Steam Generators are unavailable and the RCS is significantly supersaturated (see § 3.4 and 3.5 for adequate criteria)
3-The RCS is significantly supersaturated and temperature at core outlet is high (see § 3.5 for criteria)
4-Severe Accident Guidelines are implemented

*rationale

Number 1 corresponds to situations where coremelt prevention is the main objective. Water inventory in the RCS is significant; depressurization is thus decided to allow injection through the Safety Injection which is available and thus provide core cooling through primary bleed and feed.

Numbers 2 to 4 correspond to situations which could go to coremelt. The main objectives are the prevention of High Pressure Core-melt scenarios and to decrease the pressure in the RCS to allow injection of water when possible.
N.B.: Recent studies performed by EDF on a few enveloping Severe Accident sequences resulting in high pressure core melt scenarios tend to show that the risks of early containment failure associated with such scenarios have been overestimated. Discussions are presently underway for further assessment of these results, and no decision has been made yet on the interest of a possible backfitting of these results on accident management key decisions.

As RCS water inventory is of concern, the following considerations are made:

*if \( T_{\text{core outlet}} < 355^\circ \text{C} \) and \( \Delta T_{\text{Sat}} > 10^\circ \text{C} \),
RCS water is undersaturated and RCS inventory can be considered satisfactory. If one SG is available, RCS cooldown and depressurization is possible, and no RCS depressurization is needed. If, on the contrary, no SG is available or reusable:

+ the RCS is depressurized when the SI system is available, thus allowing adequate core cooling through primary bleed and feed
+ the RCS is not depressurized when the SI system is not available: this allows to limit RCS inventory loss and provides more time to try and restart the SI system.

*if \( T_{\text{core outlet}} < 355^\circ \text{C} \) and \(-30^\circ \text{C} < \Delta T_{\text{Sat}} < 10^\circ \text{C} \), water in the RCS is either saturated or supersaturated, but fuel cladding heatup remains limited. Decisions are made using the above mentioned arguments

*if \( T_{\text{core outlet}} > 355^\circ \text{C} \) or \( \Delta T_{\text{Sat}} < 30^\circ \text{C} \),
water in the RCS is overheated. Core uncoverage can be considered very significant, and there is a risk of cladding (if not fuel) degradation.

When at least one steam generator is available, decay heat removal from the RCS is considered to have sufficient efficiency until the temperature at core outlet reaches 700°C. Beyond this temperature, it is considered that the situation is so degraded that the major objective is prevention of early containment failure possibly resulting from high pressure core-melt.
When no SG is available or reusable, decay heat removal through the SG system cannot be contemplated, and the main objective is preventing high pressure core-melt as above. Depressurization is thus required without any further condition on the temperature at core outlet.

All above mentioned decisions are relevant for reactor situations in which no core-melt has occurred. If, however, there is further degradation of the situation and the decision is made to leave EOPs (U1 in this case) and shift to Severe Accident Guidelines, additional actions are recommended in the Severe Accident Guidelines document (see Figure 6).

- adding water

* decision

Water is injected in the RCS when the Safety Injection system is considered available

* rationale
The main objectives are to prevent coremelt or limit coremelt progression through restoration of core or corium cooling capability, if possible before vessel melththrough in the latter case. Potential risks associated with the injection of water on materials at very high temperatures, such as steam explosion or hydrogen production are acknowledged, but quantification of the consequences of some of these risks (e.g. those associated with steam explosions) remains controversial, in particular for the case of ex-vessel scenarios. To define the presently proposed guidelines, EDF took into consideration the following elements:

1. without restoration of a corium cooling capability:

a. in-vessel core-melt progression leading to vessel melththrough cannot be prevented
b. consequently, there will be ex-vessel core-melt progression and a potential for Molten Core Concrete Interaction(MCCI)
c. MCCI could lead to delayed containment failure either through basemat melththrough or pressure buildup inside containment resulting from the production of incondensibles

2. it cannot be stated that there is no risk of in-vessel steam explosion, but there seems to be an international trend for considering that in-vessel steam explosion wouldn't likely be a threat to containment integrity

3. the issue of ex-vessel steam explosion remains controversial, and further investigation is needed, but scoping studies tend to show that for credible sequences, it is unlikely that containment integrity would be threatened.

It therefore appeared that in the absence of very conclusive elements about ex-vessel Steam Explosion, risks associated with the decision of not injecting water exceeded by far that associated with the decision of adding water.

4.5-Depressurization of and Water Injection in the Steam Generators

In case of accidents, Steam Generators can be refilled or depressurized when no undue risk to the populations is anticipated.

Discussion

Decision

Add water to the Steam Generators

Rationale

When a non isolated and non contaminated Steam Generator has no or little water supply and the level is below 0% of the narrow range, the possibility of restoring the water inventory is considered. This is done when water is available, i.e. when the inventory in the Emergency Water Storage Tank is above its low-low value. If water is not "available", actions are taken to restore an acceptable water inventory in the storage tank before injection in the Steam Generator.
N.B. Adding water to non active Steam Generators is also important to reduce the likelihood of creep induced Steam Generator tube failure, and for fission product scrubbing if Steam Generator tube failures nonetheless occurred.

Decision

Depressurize Steam Generators

Rationale

If at least one of the Steam Generators is available (water level above the wide range low-low value) and not active (no contamination previously detected), decay heat can be removed first through the condenser dump, then through the atmospheric dump if the former is not available. In the absence of any contamination, steam release to the environment has no radiological impact. In the extreme case where all Steam Generators are considered active (contaminated), the activity level in all Steam Generators is analyzed to decide whether one or more can be considered reusable (see § 3.4 and Fig. ), use of these Steam Generators being contemplated only when the U1 procedure is implemented. Deisolation of these Steam Generators is then requested, depressurization being implemented depending on the evolution of the situation in the RCS. At last, if the RCS is supersaturated, all reusable Steam Generators are deisolated even in the case where at least one Steam Generator is not contaminated.

N.B. As can be seen in Table 6, Severe Accident Guidelines require isolation of contaminated Steam Generators (evolution of the activity in the SGs being then impossible to monitor).

4.6-Containment

Decision

Restore containment integrity.

Rationale

In case of accident, there is the potential for impairing the containment function through one of the following paths

1. leak at one of the containment penetrations
2. leak on the portion of one of the safety-related systems situated outside containment
3. high activity fluid in systems used for normal operation (beyond their design requirements)
4. loss of the first (fuel cladding) and second (RCS boundary) barriers.

In any one of these situations, the operator will have to restore the confinement function through implementation of the U2 procedure.
Decision

Prevent Containment catastrophic failure and open the sand bed filter (US procedure)

B. Rationale

This is only possible when Severe Accident Guidelines are used. In such a case, there are clear signs that core-melt has occurred and is still progressing, and that there is a risk of delayed containment failure resulting from slow pressure buildup inside containment caused by the loss of decay heat removal capability to the environment and the production of incondensibles due to MCCI (if any).

The decision to open the filter is made after consideration of its compatibility with Emergency Planning implementation beyond site boundary.

V-CONCLUSION

Though not actually designed to address Severe Accident issues, current French plants exhibit significant resistance capability in case of complex sequences having the potential for leading to more or less profound melting of the core. This was made possible through optimum use of existing systems, minor modifications on plants, or implementation of specific procedures. Some challenges are not yet actually dealt with, such as the Hydrogen issue. Considering risk analyses made on similar plants, especially in the US, it was felt that the risk was low and time was available to make a well argued decision.
## Severe Accident Guidelines

<table>
<thead>
<tr>
<th>SYSTEMS</th>
<th>ACTIONS</th>
</tr>
</thead>
<tbody>
<tr>
<td>Refueling Water Storage</td>
<td>Refill, in priority with Borated Water</td>
</tr>
<tr>
<td>Storage Tank</td>
<td></td>
</tr>
<tr>
<td>Pressurizer Relief Valves</td>
<td>Open all valves</td>
</tr>
<tr>
<td>RCPs</td>
<td>Stop RCPs</td>
</tr>
<tr>
<td>Safety Injection</td>
<td>if available Maximum flow (1 train if water is limited)</td>
</tr>
<tr>
<td></td>
<td>if not available Restore SIS, then resume injection progressively</td>
</tr>
<tr>
<td>Accumulators</td>
<td>Deisolate</td>
</tr>
<tr>
<td>Spray System</td>
<td>If available Same as SIS</td>
</tr>
<tr>
<td></td>
<td>If not available Same as SIS</td>
</tr>
<tr>
<td>SGs available</td>
<td>Control SG level Maximum cooling (atmospheric or condenser dump)</td>
</tr>
<tr>
<td></td>
<td>if SG filled maintain isolation</td>
</tr>
<tr>
<td></td>
<td>If SG empty</td>
</tr>
</tbody>
</table>
|                          | . if Feed Line Break outside containment    *
|                          | * maintain isolation                         |
|                          | . if no FLB outside containment             *
|                          | * close steam Dumps                          *
|                          | * deisolate water injection                  *
|                          | * refill to restore adequate level           *
|                          | * control pressure below safety valve opening setpoint |
| SGs unavailable (SGTR)   | Confirm isolation then same as unavailable SGs |
| SGs reusable             | t < 1 day Maintain Isolation                 |
| Cold Batteries and RHR   | > 1 day and Spray available Actuate systems  |
| Annulus Filtering System | Operate one train on iodine filter           |
| U2 US                    | Implement procedures                         |
Severe Accident Management Simplified

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Introduction

In 1991, the Westinghouse Owners Group (WOG) authorized the development of Severe Accident Management Guidance (SAMG) that would be generically applicable to the majority of PWR plants employing a Westinghouse Nuclear Steam Supply System. This was part of a broad industry effort in the U.S.A., coordinated by the Nuclear Energy Institute (NEI - formerly NUMARC), to assist nuclear plant licensees in responding to the regulatory requirement\(^1\) for formalized severe accident management capabilities. The scope of the development included severe accident management guidance, computational aids, and information needs which covered two of the five elements defined by the U.S. Nuclear Regulatory Commission (NRC)\(^2\) and three of the six elements defined by NEI\(^3\) for an enhanced severe accident management capability. The WOG subsequently authorized a program for the development of training material for severe accident management that would interface with that developed by the Institute for Nuclear Power Operations (INPO)\(^4\). The WOG Severe Accident Management Guidance (SAMG) and training material was developed on a generic level much like the WOG Emergency Response Guidelines and requires the definition of plant specific equipment and parameters for development of plant specific guidance. The remainder of both the NRC and the NEI requirements were determined to be too plant specific to be developed on a generic basis.

\(^1\) The Nuclear Regulatory Commission's SECY-88-147 specified the implementation of formalized severe accident management capabilities at each plant as one of the conditions for closure of the severe accident regulatory issue.

\(^2\) The five elements defined in SECY-89-012 are: procedures to prevent core damage, guidance for mitigating core damage accidents, instrumentation, training and decision making responsibilities.

\(^3\) The six elements defined in NEI 91-04 are: severe accident management guidance, training, computational aids, information needs, decision making responsibilities, and self-evaluation.

\(^4\) INPO developed severe accident management training material covering the progression and phenomena of severe accidents.
The WOG SAMG was developed to maximize the use of other industry severe accident management activities. Principally, the EPRI Severe Accident Management Technical Basis was used as a cornerstone for the identification of severe accident phenomena and their impacts. However, during the early development of the WOG SAMG, it was realized that the guidance must cover not only the best estimate severe accident phenomena and their consequences, but also the realistic bounds on the phenomena. In this case, the NRC research activities as reported in a large number of NUREG/CR documents was used to supplement the information in the EPRI report. Additional original research was also carried out during the SAMG development to ensure comprehensive coverage of the final product.

There were several groundrules established at the beginning of the development effort that significantly affected the final product. First, the guidance was to be developed to provide information on the use of existing plant equipment (for a generic Westinghouse PWR in the U.S.A.) during a core damage accident; no new equipment (or instrumentation) was to be considered. Although some modifications to the plant make technical sense for severe accidents, a cost-benefit evaluation concludes that the costs far outweigh the benefits for these low probability core damage events. Thus, if equipment or instrumentation might not be available when needed, an alternate method was developed. Second, the SAMG was to be primarily for use by the Technical Support Center (TSC), as opposed to the control room operators. There are already overly burdensome training and examination requirements on the control room staff and increasing this burden for low probability events was judged to be counter productive. These two requirements resulted in severe accident management guidance, as opposed to procedures, that was based on evaluation of existing plant indications and implementation of strategies using existing plant equipment by properly trained TSC personnel.

The WOG SAMG was reviewed by the NRC along with the SAMG from the other two PWR Owners Groups. The NRC reviewed the WOG SAMG against the elements defined in SECY-89-012, against accepted human factors criteria, and against the results of severe accident management research sponsored by NRC. The NRC feedback indicated that the WOG SAMG is an acceptable basis for plants to develop severe accident management guidance. In addition, the WOG SAMG underwent a rigorous validation hosted by Wisconsin Electric Co. at the Point Beach plant. The feedback from the validation indicated that the WOG SAMG is extremely useable under simulated severe accident conditions and only minor enhancements were suggested to improve the useability of the SAMG.

**Development of the WOG SAMG**

The program was undertaken in a phased approach whereby a general methodology for severe accident diagnosis and strategy selection was first developed and then the guidance for diagnosing the plant conditions and choosing the most appropriate strategy was written. This approach permitted the

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accident management activities after core damage. An IPE may not identify a specific containment fission product failure mode due to lack of recovery models; those failure modes may indeed be predicted if recovery is analyzed. Lastly, it was also recognized that the IPE models are based on best estimate models and do not consider the possible bounds on the consequences of severe accident phenomena. Thus, the development of the WOG SAMG had to consider all of the recognized PWR containment fission product failure modes.

The WOG SAMG

The WOG SAMG Revision 0 provides structured severe accident management guidance that is based on reference decision making process. The structured decision making process was required to minimize training requirements and provide a product that is easy to use under the potentially high stress conditions that a severe accident would produce. The structured process includes:

1. *Diagnose plant conditions* - A symptom based approach to diagnosis was employed using only measurable plant parameters.

2. *Prioritize Response* - The symptom based parameters were prioritized based on the time available for response.

3. *Assess Equipment Availability* - The availability of equipment for the response is determined first in order to avoid delays in responding to other needs when equipment is not available for the higher priority need. A key item in this part of the process is prioritizing the recovery of equipment when it is not available.

4. *Identify and Assess Negative Impacts* - The negative impacts of implementing available equipment are next identified to determine if the negative impacts are unacceptable. A key item in this part of the process is the identification of additional actions that can mitigate the negative impacts, even when the negative impacts are not unacceptable.

5. *Determine Whether to Implement Available Equipment* - Based on a comparison of the negative impacts to the consequences of taking no action, the decision whether to implement a given strategy can be reached. A key item in this part of the process is the identification of limitations and/or special conditions that may be required during the implementation of a strategy.

6. *Determine Whether Implemented Actions Are Effective* - After the strategy is implemented, it is necessary to know if the actions are effective and if the negative impacts are still acceptable. A key item in this part is the implementation of additional mitigating actions is the negative impacts become too great [note that terminating the strategy at this point will generally not mitigate the negative impacts].

7. *Identify Long Term Concerns for Implemented Strategies* - After the strategy is implemented, there may be additional long term actions required to maintain the strategy, such as refilling tanks.
use of a consistent methodology throughout the severe accident management guidance, consistent with the control room and TSC facilities and operation.

The first step in the development of a methodology for diagnosing plant conditions during a core damage accident was the definition of the key plant conditions that would indicate a need for severe accident management activities. In order to list these key plant conditions, the goals of severe accident management had to be defined. After considerable effort, the goals for severe accident management that would form the foundation of the WOG SAMG material were defined as:

1. terminate any release of radioactivity to the environment,

2. prevent the failure of any containment fission product boundary as a result of the further progression of a core damage accident, and

3. return the plant to a controlled stable condition where containment fission product boundaries would not be threatened in the long term.

In addition, two secondary goals were defined that were worthy of consideration, but are subordinate to the primary goals of severe accident management:

4. minimize the release of radioactivity to the environment while attempting to achieve the primary goals, and

5. maximize the availability of equipment and instrumentation while attempting to achieve the primary goals.

With these overall goals in mind, the next task was to develop a list of plant conditions that would satisfy those goals. This turned out to be a rather large task since the WOG SAMG was to be a generic product that was applicable to plants operated by WOG members in the U.S.A. Many of the challenges to the containment fission product boundaries are very plant specific and the WOG member plants included large dry containments, subatmospheric containments and ice condenser containments. This presented a very diverse mix of severe accident challenges. An intensive review of the findings from the Individual Plant Examinations (IPE) for the WOG member plants concluded that all of the containment fission product failure modes identified in NUREG-1150 were identified at least one plant IPE. Additionally, it was recognized that the IPE analyses generally do not model

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5 See for example, the containment fission product boundary failure modes found in NUREG-1150 which analyzed the Zion (large dry containment), Surry (subatmospheric containment) and Sequoyah (ice condenser containment).

6 IPEs are a form of Probabilistic Risk Analysis that were required in the U.S.A. by the NRC via Generic Letter 88-20.
The WOG SAMG is contained in:

A. Two control room guidelines - one for severe accident management of fast acting sequences until the TSC is available and the other for severe accident activities once the TSC is functional
B. A diagnostic flow chart and a severe challenge status tree - these are the primary diagnostic tools for severe accident management
C. Fourteen TSC Severe Accident Management Guidelines - these provide a structured evaluation for implementing severe accident management strategies

Innovative Features

Severe Accidents can be successfully managed by knowing the status of only six (6) fundamental plant parameters:

1. Steam Generator Level,
2. RCS Pressure,
3. Core Temperature,
4. Containment Water Level,
5. Containment Pressure, and
6. Containment Hydrogen

There is no need to diagnose severe accident phenomena in order to manage the accident. The applicable accident management actions are independent of the details of the severe accident phenomena and/or the progression of severe accidents. The principle actions contained in the guidelines are:

1. Fill the steam generators to protect the SG tubes, to provide a heat sink for the RCS when core cooling is restored, and to scrub any fission products leaking from the primary side
2. Depressurize the RCS to protect the SG tubes, to enhance the RCS injection possibilities, and to prevent high pressure melt ejection
3. Inject water into the RCS to cool the core, irrespective of its location in the plant (i.e., injecting into the RCS is effective whether the core is in-vessel or ex-vessel)
4. Inject water into the containment to prevent failure of the reactor vessel, to cool any ex-vessel core debris and to prevent core concrete interactions
5. Depressurize the containment to minimize fission product leakage and to prevent containment failure.
6. Reduce the containment hydrogen concentration to prevent a hydrogen burn
The issue of instrumentation survivability was also addressed in the WOG SAMG by specifying that multiple means of measuring the 6 key parameters should be used in the diagnosis process. It is worthwhile to note that the equipment qualification envelop for pressure, temperature and radiation is not exceeded for most severe accident sequences. Thus, the available instrumentation should be useful in diagnosing severe accident conditions. However, verification of the indicated conditions using diverse instrumentation indications is strongly advised.

There were several special instrumentation issues identified during a detailed instrumentation review that was part of the SAMG development. This led to the identification of special conditions that could make some of the instrumentation indications unreliable. For example, the melting of the core exit thermocouple junctions during a severe accident would lead to potentially unreliable core exit thermocouple indications upon recovery. All of the findings have been documented and none of the special conditions leads to a situation in which the fundamental parameters cannot be reliably obtained. In two cases, containment water level above the instrument measuring range and containment hydrogen, a computational aid was developed to provide added information. The bottom line is that no new instrumentation is required to diagnose and respond to severe accident conditions.

Conclusion

Severe accident can be managed with no additional plant features. Knowledge of only six fundamental plant parameters is required to determine the appropriate response. Thus, severe accident management does not require sophisticated computer codes or expert knowledge of severe accidents.
Severe Accident Management Development Program
and Insights for Closure of the Industry Accident Management Process

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INTRODUCTION AND BRIEF HISTORY

The years following the Three Mile Island (TMI) accident have seen a steady increase in the
degree of priority being placed on the control and mitigation of severe accidents at nuclear power
plants. In the early 1980's, the Industry Degraded Core Rulemaking Program (IDCOR) was
formed by members of the nuclear industry (including ABB) to develop a comprehensive as well
as technically sound position for determining whether changes in Nuclear Regulatory Commission
(NRC) regulations were needed to reflect the potential for severe accidents. In 1985, following
the completion of the IDCOR program, the NRC issued its Severe Accident Policy Statement
mandating that all existing and future nuclear plants perform a systematic examination identifying
plant specific vulnerabilities to severe accidents, followed by a submittal to the NRC for their
review.

In 1988, the principles of an accident management program, including the control and mitigation
of a severe accident, were deemed essential elements of the severe accident closure process
described in both SECY-88-147 [1] and NRC Generic Letter 88-20 [2]. In 1989, the NRC issued
SECY-89-012 [3] outlining an approach for implementation of accident management strategies,
and in 1990 the NRC issued Supplement 2 of GL 88-20 [4]. Although these documents did not
yet establish a requirement to include accident management strategies into an IPE, they were the
strongest indication to date that staff was beginning to take a stand on severe accident
management as a preplanned method for mitigating severe accidents.

The Nuclear Energy Institute (NEI, formerly NUMARC) has long recognized that severe accident
management includes many generic issues common to the entire industry. Therefore, in
conjunction with the Electric Power Research Institute (EPRI) and the PWR Joint Owners Group

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(JOG), NEI has taken a leading role in addressing these generic issues. Originally, it was believed that a single source of reference information could be compiled, from which plant specific implementation guidance could then be developed. This information source, called the Technical Basis Reports (TBR) [6, 7], was funded by EPRI and developed in the 1990 timeframe as a compendium of all available research information pertaining to severe accident phenomenology. However, industry input to the draft TBR altered its original direction by mandating a less prescriptive document in terms of core and containment damage mitigation actions. This would allow plants with varying NSSS designs to best use the TBR report without having to justify numerous plant specific deviations.

Following the issue of the TBR, the decision was made that each NSSS vendor (through their respective owners groups) would begin to develop generic accident management guidelines (AMG’s) using the TBR and other input as reference material. Ultimately, plant specific AMG’s would be developed and implemented by each utility using the generic product as a basis. However this would only occur when the NRC finally established a formal position pertaining to closure of the accident management issue. To initiate the process, ABB was funded in 1991 by the C-E Owners Group (CEOG) to develop Generic AMG’s for use by the individual utilities having C-E NSSS designs.

Over time, as the AMG development program was progressing, it became apparent that the likelihood of seeing the now long awaited final NRC mandate outlining requirements for closure of the IPE process (including closure of severe accident management issues) was steadily decreasing. Consequently in 1991, NEI (with the help of the JOG), issued NUMARC 91-04 [5] to establish the nuclear industry position concerning closure of the IPE process. By 1994, Revision 01 of the document (now called NEI 91-04) also contained specific guidance for closure of the severe accident management issue.

Although NEI 91-04 Revision 01 was originally intended to help lead the NRC toward issuing a final Supplement to Generic Letter 88-20, it took on a new mission in late 1994 when NEI and the industry came to an agreement with the NRC, stating that NEI 91-04 Revision 04 would become the basis for final closure of the IPE process, including plant specific severe accident management issues (Section 5.0 of the document). This meant that no Supplement to GL 88-20 would be forthcoming and the industry would self regulate the overall accident management closure issue, presumably with no NRC regulatory intervention. Today, the owners groups have completed their respective generic AMG’s, and the industry is slowly progressing toward completion of their plant specific AMG programs by the preagreed date of December 1998 (or earlier, as some utilities have already committed to achieve).

The following sections of this paper are devoted to several interesting aspects of the CEOG AMG development program, along with a brief discussion of issues that will need to be addressed as the utilities interpret NEI 91-04 Revision 04 and implement AMG’s on a plant specific basis.
OVERVIEW OF THE AMG PROCESS

The CEOG generic AMG's utilize a multi-phase process to first, quantitatively diagnose the current state of a severe accident (Phase 1), qualitatively verify the diagnosis (Phase 2), implement critical actions, and finally assess the impact of those actions (Phase 3). In addition, an alternate route in the process (Restorative AMG's) allows for assimilation of information under conditions where the aforementioned multi-phase process is rendered ineffective due to lack of proper equipment or instrumentation. Calculational aids (non computerized) are overlayed onto the process when necessary to assist in the decision making processes.

The basis for the diagnostic process is the "plant condition matrix" philosophy originated in the EPRI TBR. Briefly, the diagnosis involves an assessment of the plant RCS condition and the Containment condition in a manner where the final "plant condition" is determined by one of eight possible endstates as follows:

An RCS condition of either badly damaged (BD) or ex-vessel (EX) coupled with a containment condition of either closed (CC), challenged (CH), impaired (I), or bypassed (B) yields a two-by-four matrix representing the eight potential plant conditions. Once a determination has been made as to which plant condition exists, a unique hierarchy of critical actions (called Candidate High Level Actions), along with a logic for assessing the impact of taking the action, is presented to combat/mitigate the accident. The critical actions include such activities as, RCS injection, injection into steam generators, depressurization of steam generators, RCP restart, reactor cavity flooding, containment venting, containment spray and/or containment fan coolers actuation, and depressurization of the RCS. Assessment of these actions is based on a variety of well organized information in Phase 3, including statements of caution, competing effects with other actions, throttling criteria and suggestions for concurrent actions.

CEO G GENERIC AMG VALIDATION EXERCISE

On January 9-11 1995, an AMG Confirmation (Validation) Exercise was conducted at the Northeast Utilities Millstone site to test the useability and effectiveness of the document under emergency drill conditions with a compliment of TSC and operations personnel. The exercise was the Emergency Operations Facility (EOF) at the Millstone Site with attendees including C-E personnel, NU Technical Support Center personnel, Operations Support personnel, and C-E utility participants (the opportunity should be taken to thank Northeast Utilities for hosting this exercise).

Briefly, the exercise was divided into three days of activities as follows:

Day 1 (January 9) was devoted to two main activities, a morning session of background/preparation material and an afternoon session where an initial "dry-run" exercise scenario was performed. Specifically, the morning of January 9 consisted of the following presentations: A Review of Millstone Unit No. 2 (MP2) Plant-Specific Features, CEOG Generic
AMG Confirmation Exercise Groundrules, Basic Generic AMG Overview, and Instrumentation Availability/Applications. In this exercise, the priority was placed on becoming familiar with the facility, the personnel roles and the flow of information into the exercise. For this exercise (and all subsequent exercises over the next two days) the information flow, or more specifically, the flow of plant data into the exercise was accomplished through the use of detailed data sheets distributed to key personnel at pre-determined time intervals. The goal was to have the data flow be as realistic as possible so as to promote the continuity and drama that would be vital to a successful exercise. Figure 1 and Figure 2 summarize the template used by C-E and NU to formulate strategy for "moving information" throughout the participants, as well as a basic breakdown of the various stations' responsibilities respectively.

Day 2 (January 10) was devoted to the primary long exercise (Total Loss of Feedwater Comprehensive Exercise) and a shorter exercise that was originally intended as a dry-run exercise (Medium LOCA Exercise).

Day 3 (January 11) was devoted to a final short exercise scenario (Large LOCA Exercise) and a period where task participants gathered to offer comments, recommendations and to wrap-up the exercise.

Based on the agreed integration of an agreed set of minor changes to the draft generic AMG document, C-E has recently issued the final generic accident management guidelines to the CEOG.

Insights and Lessons Learned From the Exercise:

1. Operations personnel (control room staff) should always be allowed to use the EOP's, and should never be excluded from the information "loop", even during the advanced stages of a severe accident.

2. Integration of the AMG's into the overall plant emergency plan structure will not be a trivial exercise for the utilities to accomplish on a plant specific basis.

3. Although the AMG's are a TSC document, it is advised that the operations personnel be familiar (to first order) with the AMG document, to facilitate good communication with the TSC.

4. These types of validation exercises are critical to the task of discovering the "bugs" that can inhibit proper development of plant specific AMG's.
THE RELATIONSHIP BETWEEN THE AMG'S (TECHNICAL SUPPORT CENTER) AND THE EOP'S (CONTROL ROOM OPERATIONS)

One of the most important decisions in the AMG development program was the decision to make the AMG's a Technical Support Center (TSC) document for use by TSC personnel, and not a control room based document. This philosophical decision was based on the desire to transparently integrate the AMG's into a plants emergency plan with as few perturbations as possible. Ultimately, this achieves a major goal: By simply blending the AMG's into the overall E-Plan as a support document for TSC personnel to use in their decision making processes, the control room naturally continues to operate within their EOP's during a severe accident, just as they would during any accident scenario. In essence, it was decided that the EOP's would never become a "closed document" in a severe accident scenario, simply because the accident had progressed to the point where the critical safety functions were not being satisfied. Consequently, the operations personnel (and the EOP's) would not be placed in the unnatural and very compromising position of being eliminated from the decision making loop by the TSC.

The transition of command and control from the control room to the TSC during a severe accident will be the ultimate responsibility of the individual plants' emergency site director-ESD.

The criteria for transition of command and control would likely be based on two factors:

1. The plants current Emergency Action Level
2. The type of accident in progress at the time

If the emergency site director (ESD) determines that both factors are supportive of imminent damage to the reactor core, then the control room should prepare to begin receiving formal recommendations from the TSC with the AMG's being used to aid in recovery actions.

Using these aforementioned guidelines, the transition of authority during the event will take place in a very natural fashion tempered by plant specific philosophy.

Presumably, the emergency site director will gain input from the control room and TSC to make decisions necessary to recover from a severe accident. The ESD and associated TSC staff will utilize the EOP's and AMG's (as well as the emergency response information system and control room guidance) system in order to determine the proper actions to recover from a severe accident. During a severe accident, the ESD may decide that the control room actions are proper for severe accident recovery or they may decide to inform the control room to take an alternate recovery path. Ultimately though, the final decision made by the ESD will have originated from many sources of information.

It is noteworthy that, in many cases, the TSC may be staffed at a time when the plant is not yet at the point of a severe accident. Therefore, the control room would still be in full command of the event and operating somewhere within the EOP's (possibly within the functional recovery
guidelines). At this time, the TSC would not distract the control room from executing EOP’s unless called upon by control room personnel.

At some point, when the accident progresses to the point where the ESD determines that a severe accident is in progress, both the AMG’s and control room recommendations would then be utilized to help recover from the severe accident.

One final thought regarding the issue of how AMG’s will affect control room operations and interface with the emergency operating procedures: There is concern among the utilities that the AMG process will create additional burdens on the operators. Since each control room operator is required by the NRC to take a requalification exam each year, their training schedules are already filled. The addition of any type of severe accident training would ultimately mean that training would be reduced in some other area. In addition, after many years of development by the utilities and audits by the NRC, it is believed that most plants Emergency Operating Procedures (EOP’s) are finally in a state that everyone, including the NRC, is happy with. Understandably, none of the utilities are anxious to make any changes to these procedures. Moreover, if the process is a cumbersome or prescriptive one, the utilities would likely find themselves in the undesirable position of having to add additional operator training to their current E-Plan.

It is suggested that the AMG’s be present in the control room but only as reference material for operations personnel. This will facilitate a smooth flow of information between the TSC and control room (CR). Once the TSC (along with the emergency site director) has been manned, communications become critical. From a human factors standpoint, the best communication strategy is a three-way collaborative between the control room, the TSC and the emergency site director (ESD). Based on the existing plants emergency plan, once the accident progresses to a certain point (Emergency Action Level), the ESD would assume ultimate authority regarding command decisions at the plant. In essence, although the operators are the only people legally licensed to make control manipulations, the ESD would be the only person authorized to order the control room operators to perform an action. The role of the TSC would be to provide guidance and advice to the ESD and the control room personnel.

We believe that once the ESD does take ultimate command of the accident, this does not imply that the AMG’s become the controlling document.

CLOSING COMMENTS: A CEOG PERSPECTIVE REGARDING UTILITY CLOSURE OF THE ACCIDENT MANAGEMENT ISSUE

During the next few years, utilities will take steps to satisfy the NEI Industry Initiative for Closure of Severe Accident Issues (NEI 91-04). This will include at least the following topics:

1. Writing the plant specific AMG’s which include insights gained from IPEs.
2. Implementing the AMG's into an existing emergency plan structure

3. Creating training packages, as deemed necessary, for various disciplines within the utility.

The licensee severe accident management efforts will likely to encounter the following issues:

1. How to handle the interface between the TSC (AMG's) and the Control Room (EOP's) in a manner that is both efficient and familiar to all parties

2. How to deal with the concept of instrument reliability during a severe accident

3. How to integrate lessons learned from IPE / IPEEE analyses into the plant specific AMG's

4. How to deal with specific phenomenological issues that have not, by any means, reached a consensus within the industry. Examples include RPV lower head (Ex-Vessel) coolability, Containment Hydrogen Control, and Direct Containment Heating (especially for plants with tight cavity designs)

However, the most important components of the overall process are the development and implementation of a detailed plant specific program plan that captures and integrates all of the aforementioned topics (along with many other issues already discussed). Although the four owners groups have joined with NEI to carry the industry position to Washington, it is becoming quite evident that the NRC does not plan to stand idle and allow the utilities to completely self regulate their AMG implementation programs. Based on recent discussions with the NRC, there may indeed be some level of scrutiny placed on utilities, (the nature of which is still being determined) and even though this is an industry-regulated activity, the licensees will likely be very sensitive to an NRC position when plans are made to train emergency response staff in the areas of severe accident management. Therefore, those utilities that do take the time to first outline, and then carry out the genesis of their development process through a comprehensive AMG program plan (based on NEI 91-04), will certainly realize a payback if the NRC does conduct site visits/audits sometime in the future.
Figure 2

Outline of Participant Responsibilities

Exercise Coordinator(s) (ABB/CE)

- Controls exercise
  - Starts/stops actions
  - Sets timing/clock
  - Accepts/rejects actions (if necessary)
  - Releases data to players
- Plant maintenance/on-site support
- Site/pass data collection
- Supplies "special" plant data

Operations (Control Room) (ABB/CE and MP2)

- Interface with TSC
- Executes CHLA recommended by TSC director
- Authorizes/request plant maintenance support
- Reads plant data from control panels
- Provides initial accident assessment to TSC

TSC Staff (SAWG/MP2)

- Assimilates data
- Directly accesses CFMS data (if available to TSC)
- Creates AMG data sheet
- Performs plant state diagnosis
- Interfaces with control room
- Interacts with electrical/mechanical support personnel
- Recommends actions to TSC director

TSC Director: Emergency Site Director (SAWG/MP2)

- Assimilates all data
- Interfaces with control room
- Can interact with electrical/mechanical support personnel
- Interacts with members of TSC
- Recommends actions/CHLAs to control room
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Accident Management
Measures and Strategies
in Germany

- An Overview -

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Abstract

Since 1985 GRS has been investigating the various possibilities and means to influence the progressing of accidents by accident management. Preliminary results of the German Risk Study and the Chernobyl accident in 1986 led to an increased examination of all aspects dealing with the capability to cope with severe accidents. Analyses have indicated that existing nuclear power plants have considerable safety potential even for many beyond-design-basis events. There is also sufficient grace period available to diagnose the challenged safety function and to initiate counteractions mainly by using existing systems or mobile equipment. Strategies have been developed that can be taken to prevent core damage, retain the core within the reactor vessel, maintain containment integrity and minimize off-site releases.

There is a distinction between:

- Prevention, measures to prevent a core melt
- Mitigation, measures to mitigate the consequences of core melt sequences.

With these accident management strategies the well proven "defense in depth" concept will be extended into the severe accident area. As in the design range, preventive measures are given priority over mitigation measures.

In 1986 the German RSK\(^1\) has recommended to develop accident management (AM) measures. The licensees have agreed to implement AM-measures on a voluntary basis. The AM-measures are not required to cope with design basis accidents. However, these plant-internal measures support or supplement emergency response actions in case of a severe accident.

The frame for development and implementation of accident management measures was laid down.

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\(^1\) German Reactor Safety Commission
The accident management procedures in Germany are a part of a dedicated accident management manual, separated from the operating manual. The accident management manual only deals with measures to cope with phenomena in the beyond-design-basis range. It is an extension to the safety-function-oriented section of the operating manual and thus is mainly structured on a safety-function-oriented basis. The aim is to provide accident management measures for all safety objectives.

The general concept of important AM-measures for Light Water Reactors are discussed e.g.

- Bleed and Feed Measures (PWR only)
- Hydrogen Mitigation
- Containment Venting

The status of implementation will be briefly described.

The development and implementation of accident management strategies is a continuous process and further investigation, optimization, validation and training is subject of ongoing activities.

1 Introduction

The general safety objective for the operation of a nuclear power plant is to prevent the release of significant amounts of radioactive material into the environment. The design of the safety systems of operating plants is focused on the safe and undisturbed operation and on the reliable control and mitigation of specified accidents to prevent core damage and to ensure containment retention capability.

Irrespective of the precautions taken, complex failures of human or material origin (events beyond the design basis, severe accidents), were not considered during the design of the plant.
Considerable progress with regard to severe accident phenomena has been achieved during the past ten years. PSAs and severe accident analysis have shown that existing nuclear power plants have great safety potential. For most sequences there is also sufficient grace period available to diagnose the endangered safety function and to initiate counteractions by using basically existing systems or additional equipment. Accident Management Strategies have been developed extending the management of risks into the severe accident domain. Accident management adds a further level of protection to the well established "defense in depth" concept: Prevention and mitigation of releases in the case of beyond design basis plant conditions.

- Prevention considers measures to avoid damage to the core. Due to the relatively slow development from an initiating event to major core degradation there is in principle the possibility for the plant personnel to identify and diagnose the status of the plant and to restore safety-related functions, e.g. by reactivating safety or operational or additional systems. These measures are considered to have priority over measures with mitigative character.

- Mitigation considers measures to control and minimize the consequences of core melt sequences. If measures to maintain sufficient core cooling and decay heat removal fail, core melt will start progressively. Even in this case measures to control and minimize the consequences can be initiated. The final goal is to avoid an uncontrolled and large release of fission products into the environment. This can be achieved by maintaining the integrity of the containment and by limiting release rates.

The implementation of Accident Management measures needs a proper regulatory basis. Therefore, important characteristics of the German regulatory system (chapter 2) and specifics of the dynamic process to further develop reactor safety (chapter 3) are described at first. Then the preconditions for development and implementation of accident management procedures are outlined in chapters 4 and 5. Chapter 6 summarizes key accident management strategies, chapter 7 reviews the status of implementation, and in chapter 8 an outlook on further activities is given.
Legal Basis and Nuclear Regulatory System

Relating to nuclear licensing there are relatively few legal documents. The Federal Constitution (Grundgesetz, GG) is supreme, followed by the Atomic Energy Act (Atomgesetz, AtG, /AtG/) and a few Ordinances on radiation protection and procedures. Only the Ordinance on Radiation Protection (Strahlenschutzverordnung, StrlSchV) prescribes certain limits, as for example admissible doses under normal operation or for design basis accident conditions. All the other legal documents relate to relatively general requirements and administrative procedures.

More detailed requirements have been published by the BMU: Promulgations of the BMU, Guidelines and Recommendations. From a formal point of view they are not mandatory. To assure a uniform enforcement of the law a standing committee of the atomic authorities of the Federation and the Federal States has been established (Bund-Länder-Ausschuß Atomenergie). All legal activities and promulgations of the BMU but also licensing, inspection, enforcement or evaluation of operational occurrences are discussed in this committee. Based on a common consensus of the atomic authorities of the Federation and of the Federal States, the publications of the BMU are applied in regulatory practice. The consideration of severe accidents has not been part of regulatory requirements in the past.

Licensing and supervision of NPPs in Germany is not performed by a central Government licensing agency, as for example in Japan, the USA, UK or France, but devolved to the respective Federal States. All licensing procedures, inspection and enforcement activities are performed by the nuclear authority entitled by the government of the respective Federal States. The official Licensing Authority in each State does not sustain the expertise necessary for technical review, nor does it place expert resident inspectors in the plants. For technical expertise the Federal States employ the Technical Inspection Agencies (Technische Überwachungs-Vereine, TÜVs and GRS) and other consultants, thus involving many different organizations in the licensing process (Fig. 1).

The supreme Federal Regulatory Authority, the Federal Ministry for the Environment, Nature Conservation and Nuclear Safety (Bundesministerium für Umwelt, Naturschutz und Reaktorsicherheit, BMU) supervises the Federal State Authorities, with respect to
compliance with the AtG and to expediency, including the right of instructions. Thus the State Authorities are subjected to directives of the BMU. BMU directives or statements are often based on recommendations of the Advisory Bodies, Reactor Safety Commission (Reaktor-Sicherheitskommission, RSK), Commission for Radiological Protection (Strahlenschutz-Kommission, SSK) or are based on the results of special expertise that has been prepared under BMU-contract by expert organizations, mainly GRS.

The Federal Government has decided a major revision of the Atomic Energy Act. Although the law dating from 1959 has served its purposes relatively well, it should be revised:

- to become a modern environmental protection and technical safety law and
- to improve distinctness and predictability of regulatory procedures and related judicial processes.

The intention to prescribe risk precautionary measures towards severe accidents including core melt for future plants has already been introduced into the law by an article law dated April 28th, 1994. Future plants have to be designed explicitly to cope with severe accident situations. The objective is to practically eliminate large early releases and to avoid off-site measures, such as evacuation. These measures need not be applied to existing plants. But today the Federal Government considers accident management as a part of precautionary measures required by the Atomic Energy Act - as long as they are technically feasible and can be realised with appropriate effort -.

3 Regulatory Process of Accident Management and Safety-Related Plant Modifications

A strict distinction must be made between the licensing procedure for the construction of the plant and the nuclear regulatory procedure which takes effect afterwards. In the case of the licensing procedure, the authority has to pass its judgements by taking the state of the art into account, thus following the prescriptions of the Atomic Energy Act. It will use its discretionary power for risk reduction according to the ALARA-principle. After granting the operation license, the authority primarily has to check whether the plant’s status and operation is
In line with its original design and license; although the regulatory authority will constantly check whether the plant continues to fulfill the prevention requirements also under consideration of a developing state of the art, it is not in a position where it can demand further measures to reduce risks. With the unlimited operating license the operator obtains an economical constancy warrant. Not all developments of the state of the art or of licensing practices do therefore lead to later additional requirements; this would only be the case if the new findings or the new practices led to the result that the existing plant that is operated in line with its license under the legal constancy warrant no longer fulfills the damage-prevention requirements. In any case, further requirements are limited by the constitutional principal of appropriateness of means and purposes. The application of formalized cost-benefit criteria are not considered useful. Decisions are made by case by case considerations.

One may now ask with good reason why the plants that currently are operated in Germany have constantly being upgraded to a large extent. One reason is that in the last years there has been a consensus between authorities and license holders that as a matter of operational responsibility such upgrades are performed on a voluntary basis. These upgrades are often far beyond the threshold for the damage prevention requirements and do not prove that the previous solution might have been shown to be insufficient by the improved technical solution. In this context, utilities have also decided to introduce accident management measures and the corresponding plant modifications on a voluntary basis. Currently, this carefully balanced polarity between constancy warrant and voluntary dynamic improvement of damage prevention is endangered by political controversy on the future use of nuclear power.

As outlined above there have been no formal regulatory requirements for severe accident considerations and accident management in licensing and supervision. But there is a common, cooperative effort by authorities, expert organizations, utilities and vendors to develop and implement severe accident management.

This common effort is based on programs of the Federal Government from June and September 1986. In December 1986 the utilities have voluntarily offered to introduce accident management measures. The relevant scientific-technical background material has been
made available by investigative accident management programs (GRS as contractor of BMU) and the German Risk Study (GRS as contractor of BMFT). The Reactor Safety Commission (RSK) has performed an intensive review of the safety of all nuclear power plants. The RSK-report from November 1988 gives further guidance for development and implementation accident management measures. The utilities have supported the further development of AM-measures by specific investigations and proposals.

4 Characteristics and Structure of Accident Management Procedures

Accident management procedures extent the symptom- or safety-function-oriented approach into the area of beyond-design-basis accidents.

The respective accident management procedures for each plant are contained in a dedicated accident management manual separate from the operating manual which contains at first the safety function- and event-oriented emergency operating procedures for design-basis accidents.

Due to past restriction of regulations and licenses to the design basis area in Germany a clear distinction has been made between the safety-function-oriented section of the operating manual (accident conditions) and the accident management procedures. The following characteristics are applied to distinguish between accident management procedures and the domain to be covered by the operating manual:

- overriding of protective interlocks to allow unscheduled operation of non-safety-related systems
- deactivation of the intended functions of safety-related Instrumentation and Control-Systems (I&C)
- repair under accident conditions
- increased release of radioactive materials.
The operating manual contains a decision tree which makes it possible to choose measures corresponding to different plant states. This decision tree is oriented on safety functions. It also defines criteria for the transition to the accident management manual.

The accident management manual only deals with measures to cope with challenges in the beyond-design-basis range. It is a supplement to the safety-function-oriented section of the operating manual and thus is mainly structured on a safety-function-oriented basis. The aim is to provide accident management measures to maintain the important safety objectives.

A few accident management procedures are event-oriented and apply to specific scenarios. An example for this is the necessary reconnection to the grid after a large-scale failure of the general electricity supply.

The accident management manual only contains those regulations that are vital for the performance of the specified measures. However, due to the "unusual" kind of the required actions the accident management procedures have to be worked out and described precisely in order to ensure their successful performance. In contrast to the "information-based" safety-function-oriented section of the operating manual, the accident management manual will contain detailed step programs whenever possible.

It does not include regulations concerning the organization of the technically responsible crisis team as these are already laid down in other manuals. The regulations vary between the utilities and plant types.

5 Basic Principles for Development and Implementation of Accident Management Procedures

At present there exist no formal codes or guidelines for accident management procedures as e.g. for the operating manual (KTA 1201). However, there is a general understanding to apply these following principles for development and implementation of accident management procedures /ERV 92, KEE 93/:
- The accident management measures should not impair plant operation under normal or upset conditions, nor may they unacceptably interfere with existing procedures.
- Accident management measures take credit of all existing systems and equipment.
- The usual design criteria for safety systems, such as the single-failure criterion, are not applied.
- Accident management actions are in general considered as manual actions.
- The main-control-room area should be to a large extent the central location for diagnosis of plant state and initiation as well as monitoring of accident management actions.
- Accident management measures may be initiated after a sufficient period of time essential for diagnosis and decision-making and preparation of the measures.
- It must be possible to interrupt the accident management measures at any time.
- Any necessary equipment for initiating accident management measures must be arranged in such a way that operator errors or inadvertent initiation during normal operation are avoided.
- Normally prohibited actions on safety-related systems (e.g. defeating interlocks, overriding protective trips) should be permissible under proper control.
- In Station Blackout situations a recovery of off-site power supply can be considered after 2 hours (high reliability of European interconnected grid).

6 Examples of Important Accident Management Measures

6.1 PWR - AM-Measures

• "Bleed and Feed" Measures

The results of German Risk Study, Phase B, for PWRs indicate that about 98 % of the frequency of uncontrolled events (not coped with by designed operating and safety
systems) would lead to core melt under high pressure. Therefore special emphasis was laid on measures to depressurize the primary system. This can be achieved by secondary and primary side "bleed and feed" measures. The objectives of these measures are:

- restoring heat removal via steam generators by secondary side depressurization and injection to maintain long-term core cooling
- re-establish sufficient core cooling by depressurization of the primary side and feeding with emergency core cooling or operational systems
- if core melt is unavoidable, prevention of reactor vessel failure at high pressure by primary depressurization to maintain containment integrity.

The secondary "bleed and feed" measure consists of secondary side depressurization by opening the relief valves and use of the water content in the feedwater line and feedwater tank and maintaining long-term feed with mobile pumps. For better feed performance the pressure in the feedwater tank could be increased by loading the tank via an auxiliary steam line. The water volume available in the feedwater lines and in the tank can maintain feedwater supply for more than two hours. Independent feeding with external pumps can be performed using on-site water supplies (demineralized water tanks, cooling tower ponds) and external water supplies (e.g. river, tank truck, drinking-water supplies).

Primary "bleed and feed" measures consist of opening the pressurizer valves (bleed) to release the residual heat into the containment and to enable feeding with safety injection pumps or make up and additional borating system. In the long term the core can be cooled by operation of the low-pressure residual-heat-removal systems.

In case of a total loss of feedwater supply the "bleed and feed" measures will be prepared at very low water level at the secondary side. The secondary "bleed and feed" measure has to be performed with highest priority because no fission product barrier has to be given up. If secondary bleed is initiated before the break of the rupture disk in the pressurizer relief tank, contamination of the containment can be avoided. Only if the necessary actions fail and certain plant states (e.g. low water level in the reactor pressure vessel or temperature at the core outlet higher than 400 °C) are reached, has primary bleed to be performed.
The secondary and primary "bleed and feed" measures require the following plant modifications:

- modifications in the reactor protection system to enable defeating interlocks or overriding protective trips
- modifications which permit a depressurization and feeding of the steam generators from the feedwater tank
- the installation of additional connections for mobile pumps to the emergency feedwater system
- the installation of a water level probe in the upper plenum of the reactor pressure vessel
- the design of the pressurizer valves and the associated control valves for water flow conditions
- possibility for manual activation of pressurizer relief valves and the safety valves.

Hydrogen Mitigation

A large amount of hydrogen is expected to be released within a large dry containment of a PWR after the onset of an accident, leading to core damage and melting of the fuel material. Calculations of the hydrogen generation due to metal-water reactions in the core region and the reactor cavity (melt-concrete interaction) and the local and global gas distribution have revealed, that the hydrogen concentration can increase locally to an extent that poses a threat to the integrity of the containment.

Several concepts have been investigated to mitigate the possible consequences of hydrogen combustion in the containment e.g. by the use of deliberate ignition and/or catalytic removal of hydrogen. In Germany an application of catalytic removal and deliberate ignition has been further pursued.

Last year the implementation of catalytic recombiners was recommended by the Reactor Safety Commission. The catalytic recombiners are an unequivocal safety-related measure to control hydrogen.
For the catalytic recombiners, concepts of prototypes are existing which are technically well developed and proven by tests. Recombiners are passive constructional elements. Neither do they require any service by the operators nor do they require any energy supply. The installation of these recombiners within the existing PWR-plants does not pose any safety related problems.

The main features of catalytic devices are:

- Passive operation
- Catalytic recombiners reduce the H₂-concentration at extremely steam-rich and at very low H₂-concentrations
- During the reduction of H₂-concentration the composition of gas mixture is shifted towards the steam-rich corner of a ternary diagram
- Recombiners promote convection by generating heat during catalysis and thereby dissolve stratified layers
- Plates with filter jackets could protect them from the deposition of dirt and lubricants
- For filtered venting less H₂ is in the containment atmosphere due to continuous H₂-reduction by catalytic recombination
- Long-term effectiveness of the catalysts even during the radiolysis of sump water
- Continuous energy input without any burning to containment atmosphere possible

To further reduce the risk due to hydrogen combustion the feasibility to supplement the catalytic recombiners by deliberate ignition or gas dilution will be examined.

- Containment Venting

The objective of this mitigation measure is to prevent late overpressurization of the containment and subsequent uncontrolled release of fission products into the atmosphere.

The energy and mass releases into the containment during a core melt accident result in a continuous pressure increase. In case of melt contact with sump water the design pressure
of the containment can be reached after about four days in the German reference case. To maintain containment integrity a filtered venting system was installed to reduce the pressure and to limit off-site releases. The pressure relief system is designed to keep the pressure below the test pressure of the containment and to depressurize the containment to half of the design pressure within two days. The latter objective can only be achieved by cold water injection into the containment sump to reduce flashing of sump water during the depressurization.

As filter units deep bed fiber filters with an additional filter device for elemental iodine (molecular screen) or venturi scrubbers have been chosen. Both systems show a very high retention capability of > 99,99 % for aerosols. The venturi scrubber has in addition a high retention capability for elemental iodine (> 99 %). A great part of organic iodine can be retained as well.

6.2 BWR-Measures

In a large number of event sequences, hazard states can be controlled by preventive AM-measures, and damage states can be avoided. If the failure of such measures results in a damage state there are still mitigating AM-measures which can be carried out.

Preventive AM-measures are initiated if following the failure of system functions certain predicted plant states are reached. These measures are laid down in the accident management manual.

The AM-measures (examples below) serve for maintaining or re-establishing.

- sub-criticality
  - poisoning system

- RPV-feeding at high pressure
  - re-activation of the main feedwater-supply system or
  - steam driven high pressure pump
  - increased control rod drive injection
- RPV-feeding at low pressure
  - feeding with mobile pumps
  - injection with service cooling water pump
  - passive feeding using feedwater tank

- heat removal
  - increased operation of clean-up system
  - use of fuel pool cooling system*
  - containment venting

- retention of activity and integrity of the containment
  - containment venting
  - drywell spray with fire pumps or service cooling water pumps

- power supply
  - auxiliary power supply from neighbour unit or nearby hydropower station
  - coupling of diesel busbars to use any redundancy

In the following the most important Mitigation Measures are described:

- Filtered Containment Venting

The objective of this measure is to prevent overpressurization of the containment and to mitigate off-site releases.

The relief system is connected to the gas plenum of the wetwell so that the scrubbing effects of the water pool can be utilized. The venting leads via two isolation valves - it is ensured that both are closed during normal conditions - to a filter and from this via an orifice to the outlet piping. The rupture disk hermetically seals the vent system from the atmosphere and prevents the penetration of oxygen into the system. The disk will burst if the pressure in the vent system exceeds 1.5 bar. The orifice ensures a largely constant volume flow in the system independent of the initial pressure in the containment vessel by means of critical discharge flow. Thus the system pressure follows the containment pressure. Furthermore, the orifice permits a rough determination on the vented mass flow and of the activity release.
in conjunction with an activity measurement in the line beyond the filter. The check valve prevents suction of air and a possible \( \text{H}_2/\text{O}_2 \)-reaction after closure of the vent line.

The filter consists of a wet scrubber with venturi nozzles followed by a combined droplet separator and stainless-steel fiber deep-bed filter housed in a pressure vessel. The components are placed in a vessel with a diameter of 4 m and a height of 8 m.

To activate the system, the isolation valves are opened by remote control from the control room. The venting flow entering the scrubber is injected into a pool of water via a large number of submerged, short venturi nozzles.

- Inerting System

To prevent hydrogen burn and ensure the integrity of the relatively small containment vessel in BWR 69 plants even in cases of accidents with core degradation, the RSK recommended in 1986 establishing an inerting system with the following main characteristics:

- Inerting (filling with nitrogen) at the latest when continuous plant operation is achieved;
- Deinerting (refilling with air) not earlier than 24 hours before planned shutdown;
- Content of oxygen less than 4 Vol.-%;
- Temporary deinerting allowed for local inspections;
- No repumping of leakages after severe accidents.

The different areas can be supplied with nitrogen from an external source. The design fulfills the main requirements that wetwell and drywell can be supplied and ventilated independently of each other in order to ensure the specified leaktightness between these areas.

In the BWR-72 units only the wetwell is inerted.
7 Status of implementation of AM-Measures and related plant modifications

The status of implementation of AM-Measures in PWR and BWR in the year 1994 is depicted in Fig. 2 and 3. This information was provided by the Federal Office of Radiation Protection (BfS). The different status of implementation and licensing is mainly due to:

- different views of necessity and scope of AM-measures among state authorities and utilities
- different opinions between utilities and state authorities concerning the verification of the accident management measures.

8 Further activities

The following topics are subject of further investigations:

- analysis of the effectiveness and feasibility of further preventive and mitigative strategies
- probabilistic assessment of human performance for selected AM-strategies
- investigation of the use of existing systems and instrumentation outside the qualification range considering aging effects
- optimization of accident management procedures for complementary or ultimate situations
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Management Strategies
Figure 1: Organisations, their responsibilities and interactions in the licensing process
Figure 2: Status of implementation of AM-Measures (PWR)
Figure 3: Status of implementation of AM-Measures (BWR)

- AM-Manual
- Self-sufficient steam driven HPIS¹
- Diverse pressure limitation of RCS²
- Alternate injection system
- Filtered venting
- Containment inerting (only wetwell for BWR-72)
- Secured containment isolation
- Filtered air supply for control room
- Containment sampling system
- Emergency power supply from neighbour unit
- Increase of battery capacity
- Improvement of grid connection
- Underground cable (20 KV)

¹ HPIS: High Pressure Injection System
² RCS: Reactor Coolant System

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only BWR 69
not relevant for 3 single units
Swedish Regulatory Aspects on Severe Accident Management Implementation

Oddbjörn Sandervåg and Wiktor Frid
Swedish Nuclear Power Inspectorate (SKI)

presented at

THE OECD/CSNI SPECIALIST MEETING ON SEVERE ACCIDENT MANAGEMENT IMPLEMENTATION

Niantic, Connecticut, USA

12-14th June 1995

Introduction

By the end of 1988 the Reactor Accident Mitigation Program initiated in Sweden after the accident in TMI-2 was completed. Engineered measures and emergency operating procedures (EOPs), aimed at mitigating the consequences of accidents with severe core damage, had been implemented for all reactors. Documentation of the measures is given in Ref 1. The overall strategy of the engineered measures is to cool the core material in the containment and to prevent the containment failure by means of thermal barriers, flooding and, if necessary, containment venting. Releases are prevented by filtration by means of gravel bed or scrubber.

Regulatory basis

In the bill to the Swedish Parliament in 1980/81, after the TMI-2 accident, the Government proposed guidelines for the nuclear safety work within the frame of the Swedish nuclear power program. It was emphasized that, although the probability of release of large quantities of radioactive material is small, measures in order to further reduce the risk should be taken. That should be done even if the cost would not be unessential in relation to the mitigation achieved in the event of an accident. In a Government decree of 1986 the basic requirements and objectives for mitigation of radioactive releases were settled.

- Land contamination, which impedes the use of large areas for a long period, shall be prevented,

- fatalities due to acute radiation disease shall not occur,

- the specified maximum release of radioactive substance shall apply to all reactors irrespective of site and power,

- extremely improbable scenarios have not to be considered for meeting the
requirements,

In order to comply with these guidelines it was further required that any release must be limited to noble gases and at most 0.1% of the inventory of the cesium isotopes 134 and 137 contained in a reactor core of 1800 MW thermal power, assuming that other nuclides of significance in regard of land contamination are released to lesser or, at most, equal extent.

Guidelines were also provided with respect to the means of achieving the prescribed safety:

- First of all core damages should be prevented by means of high quality standards as regards daily operation and maintenance work. The on-site preparedness for accidents within the nuclear power station should be organized to take care of all possibilities to recover the core cooling, whenever accidentally lost, before severe core damages occur.

- In case of an accident with core damage there shall be prepared emergency procedures aiming at protecting the containment and reaching as soon as possible a stable final state with the damaged core properly cooled and covered with water.

- Engineered measures for protection of the containment against core melt and other loads caused in an accident should be taken so as to retain the containment function during at least the first ten to fifteen hours of a severe core melt accident.

- A controlled pressure relief of the containment as a means for accident management, though passively initiated at set limit, should be provided for in order to enable protection of the containment against overpressurization in a core melt accident.

- Furthermore the measures taken to prevent off-site consequences of core melt accidents should not have a negative impact on the overall reactor safety.

Rationale of the technical solutions

It was realized from the start of the work with the severe accident mitigation measures that there were essential uncertainties in the severe accident scenarios and phenomenology. Therefore, the solutions had to be robust both in the sense that a variety of scenarios and phenomenological uncertainties would be covered, and that new information would not make essential changes of the solutions necessary.

The main design scenario was loss of power. It was clear that the melt progression uncertainties were so large that strategies based upon retaining the core in the vessel could not fulfil the requirement of robustness. Instead, the chosen strategy was to strengthen the resistance against uncontrolled containment failure. This strategy implies that the core melt
must be retained and cooled inside the containment. Measures were taken so that the melt would fall into water and to strengthen the containment spray by introducing connection possibilities for external mobile water supply.

For the BWRs also a scenario with failing pressure suppression function was considered. The design basis for the flow through the containment venting system was that one vent pipe had broken. Since a major objective was to protect the containment integrity, the judgement was that it is better having a controlled pressure relief by release of relatively clean steam than a significant probability for failure of containment and cooling.

Venting of the containment can be initiated manually. Venting will be necessary in the long term perspective since the containment will be flooded to vessel level in the case of an unisolated loss of coolant situation. If the heat removal fails or a great amounts of non-condensibles are generated, venting will be necessary within a few hours. Since a passive function was desired, the vent lines from the containment were also equipped with rupture discs which will fail for containment pressures slightly above the design pressure. This implies that a major release of noble gases, which does not cause land contamination, is preferred as compared to a threat of containment failure.

The severe accident mitigation was not considered as a formal design basis for the plants in the same sense as the traditional design basis accidents, but it is considered as an important element that extends the defence-in-depth concept, protecting the containment integrity. Considering the uncertainty of the scenarios the systems were designed using in principle best-estimate methodology. The approach also differs from usual engineered defence-in-depth measures within the regular plant design basis in the respect that diversity and redundancy is not required to the same degree. This strategy was chosen because of the anticipated large variety of scenarios to be covered and also because of the requirement of a passive function of the systems which to some degree precludes redundancy.

**Regulatory aspects**

During the development of the severe accident mitigation program, the regulatory authority issued guidelines and directives. The review was carried out in several stages. International experience has been secured by participation in international programs and the solutions were reviewed by international expertise.

After completion of the severe accident mitigation features, the remaining regulatory issues can be divided into three parts:

- Follow up of international research to identify potential weaknesses of the solutions implemented in Sweden, and potential for improvements.

- Follow up of special remaining observation issues.

- An inspection program to verify that the implemented features will function as intended, that resources are in place to take care of a severe accident, and that
adequate training and emergency preparedness for a severe accident is maintained.

Severe accident research

Much of the research is carried out in cooperation with the Swedish nuclear industry. The objective with this research is to establish a common knowledge base. These cooperative efforts also have a historical background. The research programs which supported the technical basis for the implemented measures, the FILTRA and RAMA (Reactor Accident Mitigation Analysis) projects were conducted jointly by the Swedish nuclear safety authorities and the utilities.

After implementation of the measures, the severe accident research in Sweden continued on a reduced scale. One of the objectives of the joint program (HAFOS - Severe Accident Research Cooperation - up to 1991 and currently APRI - Accident Phenomena of Risk Importance) is to follow the international situation with regard to results and developments which may indicate needs for further improvements of the protection against uncontrolled radioactive releases or may otherwise furnish means of a more accurate assessment of the present safety situation. An important part of this has been continued participation in the USNRC CSARP (Cooperative Severe Accident Research Program) and the ACE and MACE projects managed by EPRI.

An important part of the current APRI program is concerned with the application of the present knowledge base related to severe accidents and its possible use for PSA Level 2, i.e. to systematic assessment of the safety with regard to possible releases of radioactivity in the event of severe core damage. The approach in analyzing severe accidents and assessing methods for accident management is largely centered around the MAAP code (Modular Accident Analysis Program). Accordingly, the research mainly aims at validating the MAAP4 code which is the main analytical tool used by the Swedish utilities.

In addition to the joint programs, SKI supports studies of accident progression, mainly in-vessel using the codes SCDAP/RELAP5 and APRIL. Through participation in the Nordic nuclear safety program, SKI supports code comparison programs which involve MAAP 4, SCDAP/RELAP5 and MELCOR codes. Development of computerized tools for assistance during severe accidents is also supported by the authority. SKI participated in the OECD TMI-2 projects as well as the international standard problems organized by OECD/NEA..

Issues under special observation

The focus is both on in-vessel and ex-vessel phenomena and issues. Important in-vessel phenomena and issues is the reflood of a partly degraded core and the conditions for recriticality of a partly control rod free core during reflood. The objective is to investigate a possible time window between control rod melt down and severe core damage in order to improve emergency operating procedures. Processes of corium-structure interaction in the lower plenum and probability and mode of vessel failure are also considered.
Recent findings have indicated that the probability of retaining a partly molten core in the primary vessel may be larger than earlier anticipated. Such possibilities will be scrutinized further into by the authority and the utilities.

The important ex-vessel issues are melt fragmentation and coolability in the containment, fuel-coolant interaction and structural response as well as the direct containment heating and the hydrogen issues. Installations of hydrogen control is being discussed for two of the Swedish PWRs. Experiments are being carried out at the Royal Institute of Technology to address the issue of melt coolability and fragmentation.

Surveillance of implemented mitigation measures

After the severe accident mitigation measures have been implemented the major focus from the authority side is on the inspection that the measures are maintained to work as intended, that the emergency organization is in place, and that the procedures are regularly reviewed and trained. An important aspect is that the installed measures and accident management procedures are reviewed for the specific reactors in light of the development in the severe accident research.

A special instrument for the Inspectorate are topical inspections. This type of inspection differs from the normal inspection in the sense that specific topic is reviewed, that the topical review is extended to several plants, and that the inspection team has expertise from different areas. The so-called process based regulation is often associated with this kind of methodology. Process based inspections are particularly well suited to review organizational processes, competence, and quality of organizations involved in nuclear safety.

Topical inspections of emergency preparedness have been conducted at all Swedish reactors (Ref 2). These inspections were carried out jointly by the Swedish Nuclear Power Inspectorate and the Swedish Institute for Radiation Protection. The basis for the judgements were the following criteria:

- The organization in charge should be transparent, plain and easy to understand,
- warnings and practical measure can be carried out without unnecessary delays,
- decisions are taken where the competence exists,
- regular training and development of competence take place for the personnel in the emergency organization,

Generally, the inspections revealed that the organizations fulfilled these requirements and that they would be able to adequately cope with the emergency situations.

Topical inspections have also been conducted on the EOPs (Ref 3). An overall requirement was that the procedures should be a living document. This means that the procedures should be continuously validated and reviewed both from a technical and users standpoint. The
inspections therefore concentrated on the following criteria:

- Is there a distinct responsibility for handling the EOPs?
- Are reviews and updating done in a planned and regular manner using expertise from different areas?
- Is regular training provided for the users of EOPs?
- Is there a documented feedback from the users of the EOPs?
- Is experience from other plants and international experience actively received and are there sufficient resources for development of the EOPs?

The overall impression from these inspections was that requirements on the EOPs are reasonably well fulfilled. An idea was to extend the use of full scope simulators into the severe accident area by developing new software, and to use these as validation tools for the EOPs. Such attempts have been made, but the uncertainties of the accident progression are still very significant. Since the full scope simulators normally are used for operator training, it has been explicitly stated that these facilities should not be used for training of severe accident management. Instead, the training is given as "walk-through" of essential features of a severe accident.

Summary

An overview of the legal background for the severe accident mitigation measures is reviewed. The major objective is to prevent long term land contamination and fatalities due to acute radiation disease in the case of a core melt accident. The strategy is to retain and cool the core melt within the containment and to protect the containment from failure due to overpressurization by venting. Releases which could cause land contamination are prevented by filtration. Emphasis was laid on selecting robust solutions both in the sense that a large variety of scenarios should be covered and that further modifications would not be needed.

After implementation of the severe accident mitigation measures the efforts continued at a somewhat reduced level aiming at studying risk important phenomena and following ongoing research to identify possible weaknesses in the solutions selected, handling remaining observation issues and carry out inspection programs to ensure that the organizations have viable programs installed to manage a severe accident. An instrument for this are topical inspections by which the same issues are inspected at several plants. Two such inspections have been carried out for all nuclear sites; one on emergency preparedness and one on emergency operating procedures. The inspections indicate that a severe accident can be adequately handled.

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CAMPFIRE-2000: Comprehensive Accident Management Program Featuring Innovative Research & Engineering for the Year 2000 and Beyond

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Abstract

The CAMPFIRE-2000 accident management program is being developed at the Korea Atomic Energy Research Institute synchonizing the proven state-of-the-art technologies and newly proposed innovative research and engineering. The ultimate goal of the program is to resolve the plant-specific accident management issues utilizing a coherent, consistent, pragmatic, methodical approach. The program focuses on the preventive measures to maintain reactor core geometry and the mitigative measures to secure containment integrity, should a severe accident take place in a nuclear power plant. CAMPFIRE-2000 consists of strategy assessment methods, guidance and procedures, instrumentation and information, calculational aids and tools, human and organization factors, handbook of accident management, and technical expert system. In particular, the one most immediate issue involves the simulation of the rather rapid cooling of the core debris and the reactor vessel lower head of the Three Mile Island Unit 2 nuclear plant as has recently been identified from post-accident metallurgical testing of the sample specimens. As a top-notch companion experiment for CAMPFIRE-2000, a large-scale, real-material, high-pressure system test SONATA-IV is proposed as a multilateral, multi-disciplinary project calling for international collaboration to investigate the potentially inherent, naturally-occurring in-vessel cooling mechanism from the very relevant severe accident management perspective.

1. Program Overview

A comprehensive accident management program CAMPFIRE-2000 is in the process of being developed at the Korea Atomic Energy Research Institute (KAERI) synthesizing the currently available state-of-the-art technologies in the arena and innovative research and engineering topics to be studied and applied during the next half decade. In this regard, the phased safety research programs at KAERI are listed in Figure 1.

**PHASED RESEARCH PROGRAM**

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Figure 1
The KAMPP (KAERI Accident Mitigation and Prevention Plan) comprises the preventive measures to maintain reactor core geometry and the mitigative measures to secure containment integrity. The plan is geared toward enhancing the safety of nuclear power plants (NPPs) and thus protecting the humans and the environment from nuclear hazards based on the nuclear safety principle of multi-barrier and defense-in-depth concepts. The KAMPP philosophical approach is essentially risk oriented, conglomerating the PSA spanning the plant systems analysis (level 1), containment behavior (level 2) and the consequence analysis (level 3), issue identification, scoping, planning, experiments and modeling. The NPP-specific issues (i.e. failure modes) may be identified and addressed, both separately and integrally, by roaming through germaine in-vessel (Figure 2) and ex-vessel (Figure 3) phenomena. The results are integrated, and if the issues are deemed to be resolved, appropriate preventive or mitigative options / schemes are implemented by then taking the cost benefit, engineering feasibility, diagnostics and human factors into consideration.

Figure 2

Figure 3

Thus, the overall program encompasses both probabilistic and deterministic methodologies, and is intended to cover the current and advanced design light- and heavy-water reactor plants under emer-
gency operating and accident management regimes. The program bears down upon both fundamental research needs and practical applications of this uniquely multi-faceted engineering discipline. The accident management research needs are arrayed in the form of a technology tree pursuant to procedural and phenomenological points of view of the subject matter in Figure 4. CAMPFIRE-2000 will be presented in its fuller entity at the upcoming International Conference on Probabilistic Safety Assessment Methodology and Applications (PSA'95) to be held in Seoul, Korea, November 26 through 30, 1995, principally organized and hosted by KAERI.

The accident management plans, whose framework is demonstrated in Figure 5, are currently scheduled for eighteen (18) nuclear units (14 PWRs and 4 PHWRs) in operation or under construction in Korea pursuant to (regulatory) recommendation by the Korea Institute for Nuclear Safety (KINS) over the years 1996 through 2003 (see Figure 6). The plans are basically structured into PSA and severe accident issue resolution. CAMPFIRE-2000 will conform with KINS' accident management framework consisting of five steps: 1) identify capabilities and vulnerabilities, and evaluate severe accident phenomena and their impact, 2) integrate the results from the above step, 3) develop and implement accident management program, 4) perform validation, and 5) incorporate new information.

2. Comprehensive Accident Management Philosophy

The accident management strategies of each country\(^1\text{-}^9\) have been critically reviewed in light of the recent findings of the Three Mile Island Unit 2 (TMI-2) Vessel Investigation Project\(^10\text{-}^13\), and new pathways are being sought to incorporate the lessons learned from the accident and each individual country's resolution to relevant issues. The newly identified accident management strategies will be orchestrated into a package program integrating a cohesive body of the PSA studies, the procedural and human approaches, and the phenomenological understanding.

The CAMPFIRE program consists of 1) strategy assessment methods, 2) guidance and procedures, 3) instrumentation and information, 4) calculational aids and tools, 5) human and organization factors, 6) supporting experiments and tests for the issue phenomena involved, 7) handbook of accident management, and 8) technical expert system, as shown in Figure 7.

2.1 Risk-Oriented Accident Management

The overall quantification of accident process likelihood is based upon the risk-oriented accident analysis methodology (ROAAM)\(^14\) dealing with phenomenological uncertainties in risk analysis. The logic flow diagrams are illustrated in Figures 8 and 9 for prevention and mitigation phases, respectively\(^15\). Containment integrity is the principal focus of the severe accident management program so that acute radiation to humans and longer-term environmental contamination are to be prevented. The containment failure probability is conditional on each major class of risk-significant severe accident sequences. The window for severe accident management must be defined by carefully interfacing severe accident phenomenology with PSA results. Since the containment integrity goal is limited to failure mechanisms resulting from internal accident loads, containment bypass should be addressed separately.

2.2 Software-Oriented Accident Management

A computerized operation supporting system KOSSN (KAERI Operation Supporting System for Nuclear Power Plants) was developed utilizing artificial intelligence and PSA techniques. KOSSN assists
Figure 4
the operators during an accident by displaying the present status of critical safety functions (CSFs) and by suggesting the necessary operators' actions to restore challenged CSFs up to the stage of core uncovery (PSA level 1)\(^6\). **KOSSN** may be used to support the decision making of the senior reactor operator (SRO) during an emergency. The system can be installed in the technical supporting center (TSC) so that the information from **KOSSN** can be transferred to the SRO as an alternative.

![Diagram](image)

*Supported by KAERI

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*Figure 6

The success paths are generated by using such knowledge as the structure description and present status of the systems. The success paths are ranked in accord with the operability which is defined as the system reliability. The overall structure of current **KOSSN** is demonstrated in Figure 10.

**KOSSN** will be extended to serve the purpose of an integrated accident management computational tool in TSC. The code will examine status points monitored by NPP process computers during a severe accident and make predictions about when core uncovery, core heatup and damage, relocation
to the lower plenum, and reactor vessel failure will take place. The status points may include such information as pressure, temperature, water level, radiation level, and so on. KOSSN will be equipped with an expert system and neural networks trained with a learning algorithm to make predictions. The learning algorithm will explore not just the plant data but also the plant model itself. Also to be developed are the analysis tools to explain network prediction changes, neural networks for failure detection and sensor validation, integration of normal operation and accident management operation monitoring, and the validation methodology. The desirable and probable paths as well as aggravated paths (by the adverse effect) will be generated; the candidate accident management strategies will be evaluated; and a best possible management strategy will be recommended at the time of a severe accident. The proposed flowchart for the extended KOSSN operation is depicted in Figure 11.

![Flowchart Diagram]

**Figure 7**

The records of industrial experience indicate that human actions or interactions are extremely important to managing (severe) accidents. Moreover, reviews of catastrophic accidents (nuclear and non-nuclear) identified serious human deficiencies in the areas of management, operations, maintenance, training, lack of safety culture, etc. Current PSA level 1 development activities for human reliability analysis (HRA) at KAERI will be extended to cover the realm of severe accident management.
Key CM : core melt, DB : design basis, DBA : design basis accident
PSA : probabilistic safety assessment, ROAM: risk oriented accident management

Figure 8

Key S : screening, CM : core melt, SA : severe accident
PSA : probabilistic safety assessment, ROAM: risk oriented accident management

Figure 9
2.3 Hardware-Oriented Accident Management

In the process of identifying NPP-specific capabilities and vulnerabilities, evaluation of severe accident phenomena and their impact has perhaps a most significant bearing upon the hardware changes for the plant. A wide spectrum of in-vessel and ex-vessel severe accident phenomenological issues (Figures 2 and 3, respectively) will be probed on a plant basis to propose key elements of the severe accident management scheme in terms of accident progression stages, containment failure modes and pertaining strategies.

Most importantly, a large-scale, well-focused experimental program SONATA-IV (Simulation Of Naturally Arrested Thermal Attack In Vessel) is being launched at KAERI in light of the recently
reported rapid cooling of the reactor vessel in the TMI-2 accident. Its first phase is aimed at identifying the potentially inherent mechanism of counter-current flow limited boiling of the coolant through the gap that can be formed between the mostly oxidic debris in the lower plenum and the metallic vessel wall when the debris has relocated into the water pool. This newly proposed mechanism, when experimentally reproduced and validated, could be counted on as part of the most critical accident management strategies to secure retention of the core melt within the lower head potentially without any immediate need to flood the reactor from the containment floor. This massive experimental program will be elaborated upon in Section 3.

2.4 Next-Generation Accident Management

Innovative research and development ideas are being brainstormed to develop the NEWCARD (Newly Engineered Working Concepts for Advanced Reactor Design) program geared toward the next-generation reactor and plant design. This program targets promoting first-of-a-kind engineering design concepts to enhance the safety of NPPs under transient and accident situations. The program will look at both the in-vessel and ex-vessel aspects of severe accident management.

3. The SONATA-IV Project

SONATA-IV distances itself from other lower head coolability tests in several major ways. First, this is the one experiment that deals with non-failure mechanism of the reactor vessel lower head. Second, this is the only experiment that will demonstrate the critical heat flux (CHF) boiling heat removal through three-dimensionally curved, irregular, narrow gaps formed between the debris and the lower head, and possible cracks and crevices within the solidified debris. This is one experiment that will utilize the real core and structural materials of UO₂, ZrO₂ and representative metals plus possible control materials. The experiment will involve a high-pressure test loop to investigate various material behavior of the lower head steel such as creep, superplasticity, etc. under dry and wet lower plenum conditions. The test items will also include representative relocation paths for the core debris to the lower plenum with water present or not. The experimental project will lastly pioneer innovative engineered safety features that will further enhance and augment in-vessel retention capability of the lower head for the next-generation reactors.

Initial efforts are under way at KAERI starting with visualization study (Figure 12) and the scoping test for CHF measurement (Figure 13) utilizing a pyrex belljar and copper hemisphere containing mineral oil being heated and convected by electrical heaters immersed in the pool (Figure 14). The idea here is to set forth a solid, sound physical ground on which the overall direction and ultimate goal of the project are to be based. Major milestones are laid out in Figure 15.

4. Concluding Remarks

A comprehensive accident management program CAMPFIRE-2000 being developed at KAERI has been briefly presented and put into perspective. The program blends the proven state-of-the-art technologies with newly proposed innovative research and engineering topics. In particular, the SONATA-IV project has germinated in the process, and is brought to immediate attention of the international severe accident and PSA community. Experimental reproduction of the TMI-2 vessel cooling will have a significant impact on future accident management strategies for light water reactor plants. Further engineered safety features might be introduced to enhance the in-vessel cooling mechanism.
Acknowledgments

The authors are grateful to Prof. T.G. Theofanous of University of California, Santa Barbara, CA, USA, Dr. R.E. Henry of Fauske & Associates, Inc., USA, Prof. M.S. Jae of Hansung University, Korea, and Drs. H.D. Kim and S.K. Sim of KAERI for their valued suggestions, discussions, and contributions during the conception period of the program.

Figure 12

Pressure: Atmospheric Temperature: ~ 100 °C
Gap: 0.5 ~ 1 mm
Depth: 0.5 ~ 1 Radius

1 Vessel: Pyrex
2 Heater: Copper
3 Coolant: Water
4 Heating fluid

Figure 13

Pressure: Atmospheric Temperature: ~ 100 °C
Gap: 0.5 ~ 1 mm
Depth: 0.5 ~ 1 Radius

1 Vessel: Pyrex
2 Heater: Copper
3 Coolant: Water
4 Heating fluid

Condenser

Steam Water
### SONATA-IV Major Milestones

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<tr>
<td><strong>Milestone</strong></td>
<td>Identification of Thermophysical, Material, Mechanical, and Chemical Characteristics of Lower Head Coolability.</td>
<td>Tests with Oxide/Metallic Debris under Low Pressure</td>
<td>Tests with Oxide/Metallic Debris under High Pressure</td>
<td>Tests with Oxide/Metallic Debris and Structure under High Pressure</td>
<td>Tests for Engineered Safety Features for Advanced Reactor</td>
<td>Integration of Experimental, Computational, Analytical Results</td>
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<td><strong>Experiment Phase</strong></td>
<td>I</td>
<td>II</td>
<td>III</td>
<td>IV</td>
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<tr>
<td><strong>Test Items</strong></td>
<td>25-50 kg (Simulant)</td>
<td>Preliminary Tests</td>
<td>In-vessel Cooling &amp; No Cooling</td>
<td>100-200 kg UO₂ + ZrO₂ + Metallic Materials</td>
<td>Integral Large-scale Tests</td>
<td>In-vessel Cooling &amp; No Cooling</td>
<td>Natural Cooling</td>
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Figure 15
SESSION II

SEVERE ACCIDENT MANAGEMENT

IMPLEMENTATION
Per Bystedt addresses SAMI Specialist Meeting Participants
SESAM
CSNI Specialist Meeting on
SEVERE ACCIDENT MANAGEMENT IMPLEMENTATION
12th-14th JUNE 1995 - NANTICOKE, Connecticut, USA.

EDF's experience
in the Implementation of Severe Accident Management Provisions

by

G. SERVIERE
Vice President, Nuclear Safety
Nuclear Power Plant Operations
TABLE OF CONTENT

- INTRODUCTION AND BACKGROUND
- TECHNICAL PROVISIONS and PROCEDURES
- NATIONAL CRISIS ORGANIZATION
- ON SITE OPERATIONS
- EXERCISES
- OTHER LESSONS
- CONCLUSIONS
I. INTRODUCTION and BACKGROUND

EDF owns and operate 54 Nuclear Power Plants, all of them being Pressurized Water Reactors (PWR) plus 1 sodium cooled fast breeder under specific status. EdF and the French Authorities have for long been actively involved in enhancing Safety in the area of Severe Accidents.

Severe Accident Management may have different meaning according to country or organization. Although it does not correspond exactly to the definition used in France, the paper will cover what comes beyond the design basis accidents. In the French context, this will include the so-called H procedures and associated provisions defined in order to still avoid severe degradation of the core, as well as the so-called U procedures and associated provisions, generally designed to cope with and mitigate the consequences of severe degradation of the core.

In a first step, the paper will present what are the provisions which have been effectively designed and implemented, in the terms of hardware, procedures and organization. In some cases, it will attempt to show what provisions may be linked to the specific characteristics of EdF, with 54 units belonging to a unique organization.

In a second step, the paper will try and present the lessons learned and eventually the reorientations which have been drawn from the implementation on the field in the various areas.

II. TECHNICAL SAFETY PROVISIONS and PROCEDURES FOR SEVERE ACCIDENT MANAGEMENT

II.1. GENERAL SAFETY APPROACH WITH RESPECT TO SEVERE ACCIDENTS

Nuclear reactor safety is based on providing a series of leaktight barriers between the environment and dangerous products. These are constantly monitored and are equipped with safety systems enabling operating parameters to be brought back within authorized limits. Such safety and safeguard systems must meet stringent reliability and redundancy criteria.

Secondly, safety is based on recognizing operating conditions such a normal, incident or accident transients. The facility must be designed to withstand such transients with wide safety margins, thereby precluding the possibility of unacceptable consequences for the public. These incidents or accidents are covered by Incident and Accident procedures (I and A).
Accidents considered as being the most serious are used to define and design safeguard functions such as safety injection, containment spray systems and the containment building itself, which are used in those I and A procedures.

Next, the event of highly improbable events, not taken into consideration at the initial design stage (total loss of a redundant safeguard, for example) was studied to determine the type of remedial action at operator's disposal to limit consequences and maintain safe conditions. In France, these are known as complementary procedures or H procedures.

If, in spite of everything, a serious accident should occur, such as core melt-down, it is essential that the operator still dispose of ways of limiting the consequences to the environment. This is the purpose of Ultimate procedures (or U procedures), intended to restrict releases and if possible regain control of the situation. For instance, the procedure known as U5 and the sand bed filter restrict hypothetical pressure increase and damage for the containment by discharge of the containment atmosphere, after filtering by at least a factor of 10 to 100.

As a general rule, the plant operator is responsible for applying accident procedures. When more serious accidents occur, the assistance of local and national crisis teams is provided within the Internal Emergency Plan (PUI) on-site and with the public authorities. The latter implement the External Emergency Plan (PPI) which applies to measures taken off-site.

A "severe accident guideline" in the event of a serious accident involving core meltdown has also been drawn up for crisis teams. It sets forth the most appropriate actions both for cooling the core and maintaining the containment building.

Finally, two domains are currently implemented to complement the procedures:

- A diagnosis prognosis approach in an accidental situation to appraise the situation and estimate the possible developments and potential releases in the next hours. Simplified methods and tools have been developed for the crisis team based on the state of the barriers and their support systems.

- A methodology to improve the detection at an early stage of defect of containment, in terms of monitoring and mitigation.

The general aim is to maintain radioactive release within limits in terms of both quantity and time so that specific intervention plans can be correctly implemented to protect both the public and the environment. Such release should be kept within a few thousands of the core inventory with respect to iodine for a period of about 1 day after the accident.
II.2. INCIDENT AND ACCIDENT PROCEDURES AND DEFENCE IN-DEPTH

2.1. Current procedures

Operating procedures set out the actions to be taken by the operating staff under all foreseeable incident and accident conditions.

Plant operating staff are given simulator training each year. The procedures and equipment in the control room were designed to avoid risk of error. In particular, a computerized "safety panel" helps the operating staff diagnose problems and apply procedures. Initially, the only actions required are to check that the automatic safety systems are operating correctly. Moreover, in case of incident or accident, a specially trained Safety Engineer monitors the overall state of the reactor, containment, and safety systems, using a procedure (SPI/SPE) which is different from that used by the operating staff. He can prescribe additional measures if required. This process thus provides a dual analytical level so as to avoid risks of error (NB: in the new EDF shift team organization, the role of supervisor for the procedure is played by the Operations Chief Technical Deputy; the application of SPI/SPE is ensured first by the Operations Chief and then by the Safety Engineer who is on call).

Currently most of the post-accident procedures available to the operator are based on a sequential analysis of accidents, using an initial diagnosis of the cause of the accident. Both the contents and form of these so-called "event-based" procedures have been improved over time since TMI. These procedures include:

- operating rules presenting diagnosis and action flow charts, with corresponding explanations and justifications;

- the instructions themselves which actually organize the work, with a coordinating document for the shift supervisor and manoeuvrer sheets for the operator, his assistant and auxiliary operators.

Current incident and accident procedures for plants in operation can be divided into four packages:

a. I and A procedures

These concern envelope incidents and accidents respectively used in the design and dimensioning of installations. These are LOCA's secondary pipe breaks, steam generator tube breaks and loss of power, etc. The last improvements include intermediate events (small breaks, realistic assumptions and initial conditions, etc.) and extension to shutdown states.
b. *H procedures*

H procedures deal with complementary operating conditions, essentially total loss of a redundant function. In some case, complementary means have been implemented if necessary, such as the ATWS mitigating system in case of loss of feedwater and the emergency turbogenerator group (LLS) in case of total loss of emergency power.

The following procedures have been implemented:

- **H<sub>1</sub> - Total loss of heat sink or of the systems ensuring heat transfert:**
  
  Cooling is ensured by steam generators fed by emergency feedwater system, cold water is injected at the primary pumps seals, thus permitting boration of the primary circuit and monitoring of the pressurizer level.

- **H<sub>2</sub> - Total loss of steam generator feedwater:**
  
  The hypothesis is a total loss of both normal and emergency redundant feedwater system. After verification, the opening of the pressurizer discharge valve (highly reliable pilot valves, with isolation valve) combined with starting of safety injection can provide adequate cooling by feed and bleed.

- **H<sub>3</sub> - Total loss of emergency power supply:**
  
  The hypothesis is the loss of external electric supply and loss of redundant emergency diesel generators, or the loss of both redundant electric emergency boards or bushars. An emergency turboalternator group run by secondary steam can supply electric power to IC system and to the hydraulic test pump, which can in turn ensure water injection at the primary pumps seals and supply make up water in the primary system. Cooling is ensured by steam generators fed by the turbine driven pumps of emergency feedwater system.

- **H<sub>4</sub> - Loss of low pressure safety injection or containment spray system:**
  
  In the long term after a LOCA, a single pump is sufficient to ensure safety injection and cooling. Provisions have been made, through mobile sleeves and connections to the piping, to have a backup of safety injection and cooling through containment spray heat exchangers. In case of total loss of pumping resources, provision has also been made for the installation of a mobile pump and also a mobile exchanger (U3 procedure).
The corresponding measures were submitted to the Safety Authorities by EDF and

designed to prevent or minimize the consequences of serious accidents.

Since 1981, Electricité de France (EDF) has been studying ultimate procedures

(1) Ultimate procedures

with all available steam generators and opening of pressurizer relief valves (feed and
bleed), under all possible steam generator and auxiliary systems, low and high
inertion. The 1-loop action include maximum safety injection flow, maximum cooling
and cooldown until it becomes core damage and withdrawal is possible. The purpose is to
eventually irreversible procedures applied to the core and containment building from reflooding when the
NSS and protect the core and containment building from reflooding when the

The 1-loop procedure is designed to provide the best possible coolant conditions for the

Ultimate continuous monitoring

making the decision to abandon the procedure being applied when conditions

However, abandoning the procedure being applied,

order the operator to perform limited complimentary actions, without,

which actions already demanded by the event-based procedure being applied,

monitoring operations or possibly providing the operator with continuation of the

Based on accurate status criteria, if permits:

"Safety net" for the event-based procedures.

by the same, If necessary, the ultimate procedure 1-loop is implemented. If is used as a

The procedure entitled "Permanent Monitoring Following Incident" (SPI), based on the

methodological redundancy,

available resources before damage to fuel occurs, as well as setting up both human and

independant analysis of NSSS cooling system. Their purpose is the implementation of all
covered by L and H procedures (multiple failures, etc.) They are based on an

These procedures concern all conditions not provided for in the accident scenarios

C: SPI - SP1 - SP2

都是接受的。
Exchange 5 days after a LOCA for example.

If provide for pre-heat accessible connections for operating module pumps and

Containment Spray Systems over the Long-Term Phase

Procedure U3: Use of module resources to back-up the Safety Injection and

measurement above the design pressure.

existing containment margin. Actions are planned to increase the range of pressure

Voluntary actions could be taken to delay the opening of the containment gates by

national level.

The decision to use the alternative venting system would be submitted to a High Level

bumping of the mixtures to waste containment. This is achieved through a

the stack. A pre-heating device of the piping has been installed to avoid hydrocarbon

direct (90% efficiency) through which waste passes before discharge through

inside the containment in a containment vessel system. A recirculation system and a sand bed

prerogatives in the containment vessel which is fitted with a metal preheater (90% efficiency)

porousization in the containment vessel with flue gas. If exposure planning is achieved through a

contourline and of the emergency planning. This is achieved through a

and to permit the implementation of both on-site accident management

provides a decrease in waste by removing 10 to 15% of the total radioactivity level

procedure permits containment depressurization to be carried out after 4 hours and

is reached between 9 and 12 hours. The ultimate pressure in containment vessels is levels beyond design pressure (5 bars - it is estimated

If pressure in containment vessels is to levels beyond design pressure (5 bars - it is estimated.

Procedure U5: Reducing pressure in the containment

Procure U6: Containment depressurization (except in safeguard systems)

unintended radioactive products into the atmosphere, in the hypothesis of a meltdown of the

The procedure contains provisions for eliminating early release through the ground of

Procedure U4: Early Release through Foundation Ring

(except in containment depressurization, except on safeguard systems).
Since 1981 improvements based on the state-oriented approach were incorporated into the so-called "event-based" procedures. All physically possible and nonphysical state situations to be taken into consideration.

With the state-oriented approach, a direct relationship is established continuously between the state variables of events leading to this state. The studies have shown that, whereas the number of possible accident sequences can be high, the prior causes of events leading to this state and the consequences of the correct procedure extremely difficult.

Multiplying the prior-studied sequences and the number of procedures would make in each situation the state-oriented procedures impractical.

These limitations were highlighted in the 1979 TMI-2 accident and therefore the interest for a new concept of accident sequences was increased. The difficulty to take account of the successive failures into account: incidents other than the accident, the risk to adopt a wrong procedure, and the risk of poor initial diagnosis due to human error, probe or recorder failure, etc. and the risk to commit the consequences of events.

This approach has several limitations:

The event-based approach is based on a diagnosis and analysis of a specific accident event and development of the physical state-oriented approach.
The operational form of these procedures (tests and logic diagrams) is based on logical sequences of events and incorporates a time factor for the normal sequence of events.

Retravel Resealation utilizing core prevention, heat removal to fall back, suppression and containment (primary, secondary, and heat exchangers), according to the state of the plant: stabilization, off or on.

Immediate action for each condition: Reactivity, primary and secondary water inventory, establishment of functional objectives with priorities for means of objectives of taking action. The following principles were used to draw up state-oriented procedures.

Instrumentation:

Secondary cooling system, containment and support systems, and the associated combinations, the parameters used for the various basic systems (reactor cooling system, containment, etc.).

Studies have been performed to identify the states of the NSSS and their possible progression of a measured response during the normal boiling regime to provide a basis to optimize the actions in a more effective and efficient manner. If filtering a complementary information on the NSSS state especially at a sensor to measure the water level in the reactor to provide accurate data on the status of the reactor core.

This approach is post-accident operation has required the qualification and the implementation of the long-term. EDG is developing a general implementation of the state-oriented and the L1 procedure.

and the L1 procedure.

Procedures between even-based procedures, continuous monitoring, following indicators essential operating information of the initial diagrams of expected developments (cross correlation and cross correlation, which combine or redefine separate operating criteria on main safeguard systems from the sequence of events by their purpose as follows.

The procedures were:

- Separate operating criteria on main safeguard systems from the sequence of events by their purpose as follows.
The application of state-orientated procedures and to evaluate the human factor and the operating team behaviour for specific cases are performed to validate the human factor and the operating team behaviour for specific cases.

- the allowance for multiple failures or combined events
- the process is self-correcting through closed loops (anti-stress procedures)
- the operator is not left without an answer, which is important to avoid inappropriate actions
- use by the operators ...

The tests on simulators have confirmed the efficiency of these procedures and their convenience.

...and in any situation of multiple failures and to propose available subdisplays means of guidance in any situation of multiple failures and to propose available subdisplays.

Also a number of logic diagrams enable systems monitoring to be performed.

- enable him to re-evaluate the diagnosis
- points for conditioning
- serious condition. These procedures are applied by operators and supervisors, with key
- system operating procedures and containment procedures in a more a diagnosis of the overall physical status of the NSS. Through signs diagnosis and closed
- seven interconnected reactor coolant procedures with cross-references to two secondary

The structure of the document includes the following:
This methodology will be applied by the local crisis team:

- Allowing activity in the containment to detect leaks:
  - Specific environmental measures at an early stage of the accident, even with a small amount of radiation.
  - Hypothetical leaks or containment.

In addition, a general methodology is under development to cope as early as possible with simplified approaches and without worsening the situation.

This guideline will be applied by the crisis team to help real-time decision-making process. It is intended to give the main strategies based on the best estimate current understanding in a simplified approach without worsening the current situation.

The main actions are containment spray (reduction of containment pressure), isolation, and cold shutdown (which are characterizing of core meltdowns). The plant is designed to be applied beyond the domain of the accident and to bring back the plant under control.

A general guideline has been established to cope as best as possible with severe accidents.
Restrict repercussions which could be damaging to the public and the environment.

Public authorities are advised by these technical organizations. They have the responsibility to inform the public about the emergency. The local and national levels, through a specific telecommunications network, allow a dialogue to be established with the teams at the plant.

The Internal Emergency Plan also provides for informing public authorities at both local and national levels. Through a specific telecommunications network, it allows a dialogue to be established with the teams at the plant.

Following the accident, the purpose is to control and restrict as much as possible any repercussions. The main items are evacuation of the public inside the plant and of the surrounding area. The Internal Emergency Plan provides for immediate deployment of the plant on-call teams.

In the event of a serious accident, an Internal Emergency Plan will be triggered by the plant.

III. NATIONAL CRISIS ORGANIZATION

Information for the implementation of special cyber intervention plans.

The objective of this work is to be able to estimate the likely time limits before core meltdown.

EDF is currently implementing methodologies and tools to conduct diagnostic and pre-accident estimation steps under accidental conditions, primarily breach size, confinement pressure behavior, source term and releases in the environment. The main items are calculation of the consequences, evaluation of the barriers and system stability. The main items are calculation of the consequences, evaluation of the barriers and system stability. The main items are calculation of the consequences, evaluation of the barriers and system stability.
The organization includes national authorities with responsibility for nuclear safety (DSIN) established by governmental order.

The organization of public authorities in the event of a nuclear incident or accident is

Coordination between EDF organization and public authorities

than a few hours.

In case of emergency, all those people, whether on call or not, have to check in with different managed through the procedures by the local team in the control room.

management team the procedure by the local team in the control room.

A crisis organization from Framatome (contractor for the nuclear island) may also be put into operation. It comprises special skills in the behavior of the NSSS under accidental conditions.

Specialists may be sent to the site to assist the work of the local crisis team.

meteological conditions

release into the atmosphere and the possible repercussion in the proximity given expected consequences into the behavior of the NSSS under accidental conditions. They can therefore estimate the possible risk of waste management and environmental contamination. They can on-site analyze the expected status of the installation and recommend, after a nuclear accident, initial experts of the situation and accident reports, the EDF's organization includes specialists in the various areas affected by the accident.

In addition, notably a crisis director, a security officer, a technical officer, and a speaker,

This team is composed of a minimum of 5 people plus assistance for communications. It

EDF's National Organization

includes 12 teams in special equipped offices: Management at national level and coordination information given to the public by the Plant

EDF's organization includes a system whereby the following personnels can immediately be summoned to meet in places in specially equipped offices:
The Nuclear Installations Safety Directorate (DSIN) has set up a specific organization to master nuclear accidents.

This comprises the following:

- a management team led by the DSIN headquarters,

- a national expert crisis team comprising engineers from the CEA's Institute of Nuclear Safety and Protection (IPSN).

This team is in constant contact with EDF specialists, both with national and local crisis team. It is linked with the computerized safety panel. It appraises the situation and any possible developments, and keeps the DSIN headquarters informed.

In terms of public radiological protection, the Local Public Authorities (the Prefect) make decisions based on data supplied by the plant and the related ministerial bodies, especially the Health Ministry - office for protection against ionizing radiation (OPRI).

The Industry Ministry is responsible for coordinating the national plan for informing the public and the media.

c. Communication resources

Coordination between the various teams mentioned above requires that they have the pertinent information at their disposal simultaneously and that they can communicate with each other easily.

Setting off the alert

The EDF crisis team comprising on-call engineers is alerted by "Eurosignal" or equivalent means. The plant issues an initial description of the incident which is then analyzed by the engineers. The emergency system established by Framatome and the Safety Authorities are similar to those of EDF.
Finally, a national team of emergency controlled robots has been constituted by both EDF and the French group to operate monitoring equipment which could be required under accidental condition produced by accidents.

Ventilation system incorporating silencers will be provided by a special provider, allowing access to the site. Each function in the control room will be provided by a special provider, allowing access to the site. Each function in the control room will be provided by a special vehicle, and there will be special vehicles of protection systems provided.

Effective of protection systems provided.

Complementary studies have been performed to accurately apprise the operating possibilities. Complementary studies have been performed to accurately apprise the operating possibilities.

ON-SITE COMMUNICATION OR RADIOACTIVITY

IV. ON-SITE OPERATIONS UNDER SERIOUS ACCIDENT CONDITIONS

and the various bodies concerned.

and the various bodies concerned.

Communication networks have been established to facilitate communications between the plant and the crisis office.

and the various bodies concerned.

and the various bodies concerned.

Direct exchange of analysis between crisis and management teams.

During a second stage, a data processing unit (KTI) is established with the unit All images and safety system operation.

During the first stage, engineers and crisis teams receive data and displays from the data processing to status of the unit in which the accident has occurred are transmitted to the crisis teams autonomously.
Valuation of which is unknown.

of questions and present and by journalists who may have other sources of information, the
public information procedure, this includes the training of EPR officers to deal with the type
Journals may be invited to participate in these exercises to check the correct operation of
responsibility for protection of the environment and safety of people.
participants. An extremely detailed scenario is employed to test the responses of persons
an annual exercise in which the entire crisis organization of both EPR and the public authorities
Finally, the General Secretary of the International Committee for Nuclear Safety organizes

for mishandling and learning.

purpose of this second phase is to evaluate the ability of crisis teams to devise remedial action
Numa numerical analysis leading to the potential consequences. The
phase involving a simulator and a second phase involving a scenario programmed to produce
The DSN organizes two of these exercises per year. Usually these exercises include a first

seven exercises are carried out per year.

Each of these exercises in an exercise of this type every two years. On a national level, six of

Simulated conditions are complex and include a combination of cumulative, multiple, incidents
on the hypothetical Caithness accident code.

Case study simulations of which the real time computer simulation is based
Naturally, these exercises are based on accident scenario simulations using the Training


The role of these exercises is also to ensure appropriate personal training.

Exercises on all planned aspects of the organization's ability to control accident conditions.

V. EXERCISES
Phrase:
In certain situations the importance of logistics, especially in the field of communication, means

Knowledge:

A number of EDF engineers

Another evolution defining at least partly from this need of continuity and completeness is

One application, amongst others, was the development of the H procedures. It has been said

The first point in the necessary of having a set of procedures or guides covering a range of

Nevertheless let me try and summarize some of them:

Evolutions which took place.

It is therefore difficult to identify lessons learned differently than just by describing the

Process as well as by lessons learned from external (to EDF) events.

The lessons learned have been recorded and documented by the lessons learned all along the

In the previous description it can be seen how the historical and progressive development of

VII. OTHER LESSONS LEARNED.
Plan:

By the national team and not the country. We must not end with a remote operation of the national team.

Operational decisions must be kept at the site, even though the site is helped and supported at the appropriate level and according to their deployed roles.

Successful national decision making in the technical arena associated by the fact that the national teams are more often trained and sometimes lead the local teams to pass the hand to the national teams a little too much during the exercise. Therefore, there are many interventions in favor of a strong national level.

The integration of a national organization of EDX as well as its size and the relation with the local and national responsibilities in crisis teams.

Local and national responsibilities in crisis teams.

Teams and their actions.

In the event that one of the bridges of measures, designed to improve the operation, fails, they are one of the best ways of achieving success, but they are also the measures that are often damaged or destroyed.

Such responsibilities need time to be set into place, especially in such a large and integrated system.

An effect of the same nature can be sometimes seen within the national crisis organization.

Therefore, the decision to modify the organization to introduce an additional engineer and head of the operational team within the after effect would be a well-received one. The aim is to give back all the operational responsibilities including safety to the normal management line.

These engineers (TSR in French) are to be quite efficient and useful, real redundancy and surveillance ultimate processes.

Safety Engineer

From team to team, can be noted also.

A few other points may be more questionable because more difficult from site to site or even
assumptions

In this instance, it is important that the necessary changes in the area not every situation can be evidenced. It is important that people know each other and how to work together. Rather than know what to do.

Necessary to deal with all aspects, particularly with organizational and training aspects.

Technical background is available and is shared by designers, operators, and regulators.

Necessary that implementation of provisions is only decided when sufficient scientific or interesting results are to be expected.

Avoid unnecessary complexity or requirement in procedures even if it may appear necessary or being preferable.

Years are as follows:

The main lessons which can be drawn from this effort that lasted already for more than 15 years are the following:

Of course, this development has been progressive. Some may even say if this has been slow.

Teaching procedures as well as organizational provisions.

EDP developed a comprehensive and coherent set of provisions which include hardware.

VI. CONCLUSION
JUNE 13, 1995

IMPLEMENTATION
SEVERE ACCIDENT MANAGEMENT
ON
OECD SPECIALIST MEETING

Carolina Power & Light Company

Director, NEI Regulatory and INFO Affairs
Frederick A. Emerson

CAROLINA POWER & LIGHT COMPANY
AT
ACCIDENT MANAGEMENT IMPLEMENTATION
available for utility use in early 1996.

reserved by the BWR-4 Mark I Brunswick Plan. Because CPAL operates two different two-loop Westinghouse plans (H.B. Robinson and Shawron Plants) and the dual-unit Carolina Power & Light Company (CPAL) supports the industry initiative, CPAL operates

balance the expenditures of company resources appropriately across all four plants.

and ensure group products provided for use

Give due consideration to the value and cost of implementing the specific NEI, INPO,

the development of new programs and procedures

make full credit for existing operating and emergency planning practices and minimize

is not excessive. Given the low risk significance in this area;

ensure that the degree of implementation reflects NEI guidance and NRC positions, but

Vendors' implementation must be managed carefully in order to

BWR-4 Mark I Brunswick Plan. Because CPAL operates four plants with two different

W

last, for each plant.

compliance, or a commitment to implement the NEI position with a schedule to be provided. NEI requested that each utility comply with the NEI position with a schedule for December 31, 1993.

Each Licensee will implement the Performance Guideline, i.e., in accordance with the SEAC Issue and may add sections.

Assess current capabilities to respond to severe accident conditions using Section 5

December 31, 1996. The full text of the initiative is as follows:

utilized to implement a severe accident management (SAM) program at each plant by

Committee will monitor implementation in terms of improving industry initiative which contains each.

On November 4, 1994, the Nuclear Energy Initiative (NEI) Nuclear Strategic Issues Advisory
Planning the self-evaluation process for the 5AM program

Upgrades to the 5AM program

Training program development

Experience planning process changes

Strategic development based on PSA reviews of WOC and BWROC strategies

Implementation of the CEP, when integrated into the development of the WOC and BWROC

Products to fulfill the implementation

Knowledge of CEP, personal involvement in the development of the WOC and BWROC

Implementation during 1996 to 1998. This phase will include

Summary

Intended to do this: ensure that the 5AM approach outlined in this paper is as effective as possible. The PSAs should be used to establish the self-evaluation process and plan.

Implementation of this severe accident management initiative will be managed through the established Project Authorization Process at each site. The overall management of the

SUMMARY
The first step is for the corporate Probabilistic Safety Assessment staff for each site to review plan PSA/PE levels relative to the WOG or BWOG set of accident management guidance.

3.2 Implementation

3.2.1 Sequence Development

Self-evaluation
Updation
Training
Emergency Planning Process changes
Plan S/A development

Implementation phase addresses the following areas:

The process described here is intended to be generic for all three sites, however, it may be modified.

Project Implementation Phase

1993
1995 and beyond. This estimation process is expected to involve a low level of effort in cost estimation for implementation. This cost estimate will be used in budget preparations for other plan projects.

Consider the expected implementation steps listed below, the group will in 1995 develop a

This group will be supported by the appropriate corporate PSA engineers, and will include a

3.1 Project Study Phase

3.0 IMPLEMENTATION

Regulatory Affairs
Nuclear Engineering
Emergency Preparedness
Training
Operations
The genetic strategies developed by the WOG and BWROG are intended to be applicable to a

Group criteria, but should not be restrictive or burdensome.

Provide brief but appropriate documentation of the rationale used for developing the

Accident Management Quicktime (PSAQ) set:

assign the initial value to one of the OPs or the Plant Specific

site (where they are most effectively managed). The PSAR will become
decision (Control Room Operations or Routine ERO).

In the case of BWROG, only, the BWROG has developed placement criteria in

address the risk-significant PSAR results at each site, or for other compelling reasons.

This result is the subject of WOGs, Group Genetic strategies which are deemed necessary to

incorporate certain low-value strategies in order to address Plant Specific issues.

implement certain low-value strategies that are deemed necessary or are deemed necessary to

assert the entire strategy is implemented properly. Also, it may be prudent to

address significant PSAR results, and thus the low value elements must be reviewed to

separate out some low-value (risk insignificant) strategy elements from those necessary

NOTE: "Other compelling reasons" refer to the fact that it may be difficult to

Review the proposed WOG or BWROG accident management strategies against the

Assignment to SAQs:

Determine which PSA results the NEI 91-04 guidelines would currently resolve through

IPE submissions to address IPE findings.

Then review the current PSA to evaluate the impact of other actions taken since the

sequences with repercussions between 1-E-5 and 1-E-6.

IPE Core Damage Evaluation Process (CDP) (using the NEI 91-04 Section 2.0).

Serious Accident Management Quicktime (SAMQ) using the NEI 91-04 Section 2.0.

Review the IPE results, including the accidents scenarios that were "assessed" to

early release as well as core damage frequency (CDF), as follows:
incorporate into training programs only those INPO and Owners Group products that are

As before, the plans will take as much credit as possible for existing training programs and

plans will be developed depending on the needs of the BWRO-operating members.

specific lesson plans and other training materials. It is possible that both a set of BWK lesson

least overall objectives, expectations, and reference lists to support development of plans. All

to develop these materials to make them plan-specific. The BWRO products will include all

The WOG training products will include lesson plans covering the strategies and are intended

Lesson plan on PSR applications
Lesson plan on damage assessment required to sever accidents
Lesson plan on procedures required to sever accidents
General risk units
Severe accident overview training (computer-based)

Programs: The INPO products include

Owners Group training products regard the current operator and Emergency Plan Training

As was described for the Owners Group strategies, the site team will review the INPO and

minimum amount of utility work to make them useful.

from the development of these products, computerized CRKL start will provide

by the schedule for the development of these products, computerized training materials to the

pumps in their plan-specific training on severe accident management. To the extent allowed

INPO and the two Owners Groups are developing generic training materials to be used by the

3.2.3 Training

of Emergency Plan changes to be implemented will be developed.

Electricity with core mental wear is added to the process.

each Owners Group to enhance these areas as the complexity of

evaluation and analysis, communications, training, and decisions making. New generic tools

assessing implementing processes already address sparsely monitored and focussed,

responding to, and protecting the public from, a severe accident. The Emergency Plan and its

Plan-specific Emergency Plans currently provide considerable capability for assessing

3.2.2 Emergency Planning Process Changes

which add little to the plant's ability to respond to a severe accident.

Each site will review those SNM process enhancements from the WOG and BWRO.
evaluations

Credit will be requested from the NRC to substantiate these self-evaluations for any NRC.

would be held to validate the strategies and processes previously developed. An initial evaluation

had been submitted from emergency planning drills and would emphasize in-plan accident

based on actual emergency exercises. This drill would be similar in format to a

scenarios development for these drills. The frequency of these self-evaluations should be on a

schedule of at least one every two years.

The site Emergency Preparedness Groups will develop a schedule of audit topics for mini-drills to

3.2.5 Self-Evaluation

industrial information as well as plan changes (as part of the periodic update of the PPS).

secret action management-revised procedures and edctions (based on revisions of new

will be assessed in the Corporate PSA Group, which will recommend to the site changes to

responsibility with those of no general industry responsibility. A CPALJ is responsible for

make appropriate program changes. It is expected that these will be primarily at a

One of the desired elements of plan severe accidents management programs is the ability to,

3.2.4 Updates

accidental management at each plant.

The result will be the subject of generic training program changes needed to implement severe

and NRC Headquarters and Region personal.

accidental management considerations in operator training and licensing and licensing utility industry.

with NRC personnel to achieve a consistent understanding of the role of severe

of SAIC is outside the realm of operator regulation. CPALJ staff will continue to work

has indicated that the real set of

The WDCO faces the challenge of SAV training and licensing in the context of other training needs and

Al Bunsiewicz, the WDCO Training Prioritization criteria will be applied to assess

neccessary to provide training on the set of strategies and E-plan changes determined from

Preparedness drills and exercises.

This self-evaluation process should be conducted within the context of the emergency

that the area is well beyond the design basis of the plant. Self-evaluation should be conducted on the role of regulatory inspection versus utility self-evaluation in this area. Given

maintenance of a plan accident management program. Differences between NRC and industry

The evaluation of accident management capabilities is an important consideration in
GKN II
Deployment of accident management measures in Germany

Nuclear Power Station Neckarwestheim (GKN II)

Report of the

OECD Specialised Meeting on Severe Accident Management Implementation

Niantic, Connecticut, USA: 12th - 14th June 1995
Minimize the effects of a beyond-design-basis accident

At the supply for the control room, the systems were provided to

determine the multi-inclusion of component malfunction qualitative and

just before the commissioning of the Brokdorf Nuclear Power Plant. It has been

construction.

resource political and public acceptance of operating plants as well as ones under

down in the Atomic Energy Act. These voluntary actions were taken to help

plant minimize the risk posed by their plants beyond the provisions of the

nuclear power plants declared their willingness to take additional measures to

Chemnitz accident. After the shock of this event, the operators of Chemnitz

system

procedures, installation of reactor vessel level measurement and valve locking

operation of German plants such as development of suppression breach

Sweden. At this time there were only minor effects on the commissioning of

measures were introduced in other countries, particularly in the USA and

TMI accident, efforts in the area of development of accident management

begin of work for the German Risk Study Phase B. As a consequence of the

contribution in the course of the Risk Study Phase B.

individual sequences were selected for closer examination of their risk

obtained procedures were incorporated into the operating manual. However

on General Operating Experience Feedback based Critical Safety Function

decisions. Selected technical improvements in the plans were made based

improvement measures since the almost finished Risk Study Phase B showed

TMI accident. This had no direct consequences with regard to accident

Reactor Safety Association (GRS)

German Risk Study Phase B of the Biblis Nuclear Power Plant by the German

development in Germany follows:

associated with the General History of accident management, a short chronology of its

Since the implementation of accident management measures at GKN II has been closely
In 1987 Siemens/KWU was commissioned to implement the following accident management
measures: at least in the external implementation in Bredkort, had to be installed:

- Sampling of the containment atmosphere.
- and emergency control room
- Isolation and access-restricted ventilation for the control room
- Access-restricted ventilation of the containment

Checklist was in the middle of construction, to be granted an operational license, accident
At the time of introduction of the first accident management measures at Bredkort, the

3. Plant Specific Developments at CKN II

Research and research establishments are carrying out the relevant basic research:

- Development of Beyond-Design-Basis Emergency Management (BDBEM) Procedures.
- Development and implementation of accident management measures.

The latter were compiled in a Beyond-Design-Basis Emergency Manual, 6th:

- Accident scenarios and those intended to cope with beyond-design-basis accidents
- Measures to clearly distinguish between measures taken following design
- particularly with the objective of avoiding core melt events (proven to be.
- recommended the examination of further accident management measures in
- the so-called 4th safety barrier, 6th. Simultaneously the RSK
- Nuclear Power Plants. The accident management measures were established as
- important and resulting from accident management in the design concept of
- An essential policy document of the Reactor Safety Commission on the
- development and assessment of the accident management measures planned at
- First official statement of the Reactor Safety Commission (RSK) on accident
- Total loss of feedwater supply
- Suction blockage and
- Certain high-pressure small-break LOCAs scenarios

core damage:

As shown in the following events make the highest contribution to the frequency of

The German Risk Study, Phase B, was used as a basis for the selection of the procedures.

- Beyond-design-basis accidents
- There were not to be numerous procedures added to specific scenarios: instead the objective
- The consequences of a core melt accident
- Or
- The probability of a core melt accident

In contrast to other trends, CHN stearey from the beginning was to limit the number of

procedures. Only such procedures were developed and included which had the potential to


The initial concept of the Beyond-Design-Basis Emergency Manual was worked out in

4. The concept of the CHN II Beyond-Design-Basis Emergency Manual

Because it followed from the above that a BEM for CHN II was to be developed

because it required in terms of the Atomic Energy Act. After the granting of the operating

measures was a voluntary act on the part of the utility. There was no legal basis for the

point of view that this would also have been difficult to implement. The concept of accident management

development of accident management processes had only just begun. From the knowledge

time it was no longer possible to generate a specification from the technical point of view at the

expansion of the BEM (by general terms without further specification) is concerned. This

is the installation of a boiling filter (crudeback) in the containment ventilation system as well as the

In the opinion of the science community in Dec 1988 the licensing authority demanded the additional
6. Implementation of Secondary Side Bleed and Feed (sec B & F)

Plan recovery after station blockout

and to be implemented soon:

primary side bleed and feed
secondary side bleed and feed
simply of the containment atmosphere
tried control room air supply
tried vents of the containment

Procedures

The Beyond Design Basis Emergency Manual at pressurized contains the following main

5. Accident Management Measures in GKN II

the primary boundary
basics scenarios and above all have the potential to prevent high pressure failure of bleed and feed were retained. They are still widely applicable in different beyond-design
overall core melt scenarios. The procedures originally chosen mainly secondary and primary
(result of the new changes in technical detail as compared to Bubis B), a power
Although preliminary results of actually performed Kovon plan-specific PSA yielded,

In choosing the accident management measures for GKN II

the reactor coolant pressure boundary? This was an additional criterion taken into consideration
Wihout appropriate countermeasures these sequences can lead to a high-pressure failure of
Isolation valves which have to be exercised in the case of a station blackout. For control units of the main steam relief valves as well as those of the concentric feed water main, primary-side bleed is to be controlled manually from electrical connections to the plant. In addition, one of the primary bleed and feed connections from which the measures are connected to a second deaerator concentric deaerator power supply. The control and isolation valves required for the secondary-side electrical management.

6.2 Modifications to Electrical Equipment

6.1 Modifications to Mechanical Equipment

Once in the atmosphere, the exhaust pipe was permanently insulated to extract the exhaust close to the nozzle. An exhaust pipe was permanently insulated to extract the exhaust in the building. For use of the mobile feed water pump and the necessary hose pipe were placed in storage. Delivery lines of the concentric feed water pump header located in the emergency feed water pump. To use the mobile pump, additional nozzles were installed on both the suction and the discharge.

When modifications were necessary, the required equipment was to a large extent already existing in the plant, and therefore only minor modifications were necessary. The required equipment was to a large extent already existing in the plant, and therefore only minor modifications were necessary.

Module pump:

- the feedwater tank pressure in case of delay in starting the back-up pressure feed through the feedwater pump.

- source pools:

- deaerated water from the concentric feedwater;

- water line and later回到 feed by a mobile pump with the first pressure feed with preheated water from the feed.

This ensures feedwater supply by two main rows to be reached.

7
Conditions

some of the modifications have any influence on designed normal and unusual plant
not initiating where with severe grade or operational instrumentation and controls. However, have the common elements of
added or existing ones modified. All modifications, however, have the common elements of
as in the case of secondary bleed and feed, mechanical and electrical components had to be

4.2. Flow conditions (AV/WS qualified).

For single-phase (steam or water) or two-phase flow conditions (AV/WS qualified).

60 cm?- have to be opened. The depressurizer Relief and safety valves are designed and qualified.
Per an effective primary bleed of 3 depressurizer discharge valves with a cross section of

prepare primary bleed and feed is shown in Fig. 1.

cooking. This section will at least avoid core melt at high pressure. The initiation criteria to
pumps so at least establish core computer in the unlikely case of failure to establish core
boreed water in the safety injection pumps, accumulators and the low pressure injection
for secondary bleed and feed. The objective of these measures is to depressurize the
secondary containment measures have been unsuccessful. They are therefore a backup measure

The primary side acidition containment measures only come into use if the secondary-side

7. Implementation of Primary-Side Bleed and Feed (PIM & F)

special procedures are then carried out by the plant personnel.

When the initiation criteria for performing secondary-side bleed and feed are met, the

6.4 Performing Secondary-Side Bleed and Feed

leck, which is initiated after reaching initiation criteria of sec. B to F.

leck, which is initiated after reaching initiation criteria of sec. B to F.

This starts the preparations but not the actual performance of secondary bleed and

appropriate manual actions are carried out in accordance with the specified procedures.

When the initiation criteria to prepare for secondary-side bleed and feed are met at 18:00, the
primary block and feedback over muffling once.

processes included. Priority is of course given to preventive measures, such as secondary or
deployment of office equipment resources such as evacuation of the local population can be
operational or safety equipment beyond the intended limits so that the necessary for
symptoms based treatment. The objective is to reduce the residual risk by the use of
accidential management procedures in the beyond-design-basis manual as an extension of the

resource them.

procedures, the operators will use beyond-design-basis accident management procedures in
10 fulfill the critical safety function with measures described in the symposium based
condition by using the symptom-based procedures. If for some reason it is no longer possible
even identification is not possible, the operator will suspend the plan to a long-term safe
nevertheless still rely on this continuously to check the critical safety functions. If a possible
condition is identified into a long-term safe condition using the extended beyond procedures.

At the time of the initiating event in the case of a positive even identification the operator will
determine the initiating event. In the event of a positive even identification the operator will
plan parameters which are initially assessed to the critical safety functions. If the safety
functions are fulfilled, the operator can continue to modify the even and proceed to

The critical safety functions are primarily examined during the even by checking a few

the beginning of the even.

plan is in a safe and stable state following the automatic initial safety measures triggered at
functions and the operational status of the safety systems he is able to determine whether the

the even is in accident is to determine the plan status. By choosing the critical safety
basis events as well as beyond-design accidents. The most important part of the operation at

Figure 12 shows in a stripped form the personnel concept of even handling, covering design
existent accidents in plant design are used and the symptom-based procedures have

combination of such events.

confounded by symptom-based procedures in order to cover unforeseen events or a
After the TMI accident the existing even-oriented emergency operating procedures were

6. Accident Identification: Event Handling Scheme
The document is completed by the relevant diagrams and tables.

Detailed instructions for on-site actions:

Modules (detailed instructions) are selected

Strategic overview with the help of which the required

Criteria for initiating the procedure

Assembled in "modules" a procedure is built around the following elements:

The document consists of two levels of detail: a loose leaf file of individual measures

Procedure (shift supervisor, control room supervisor, etc.) only the material required.

makes it possible to separate the different levels of detail and thus to give every user of the

BEM of GKN II the module structure, schematically shown in fig. 14, was chosen. This

The requirements with respect to maintainability can be met by various types of layout. For the

for core tools is high for these plan courses.

comprehensibility of the procedure because the potential

Special emphasis is put on the maintainability and

sometimes in relatively short time,

are activities to be carried out outside of the control room.

in addition to operator actions in the control room there

functions: they are not even based procedures.

The procedures are structured around critical safety

The formal structure differs, however, from the SOP inspector as

emergency operating procedure (section on strategy and even sequence).

emission of the procedure. The level of detail is comparable to a standard even operation.

self-explanatory paper which includes all necessary information for performing and

Every accident management measure within the BEM (fig. 13.1 and 13.2) is a self-contained,

Manual

9. Structure of a Procedure from the Beyond-Design-Basis Emergence
10. Licensing Procedure

Changes of procedures within the design, as any other changes of hardware or operating instructions, have to be approved by the relevant authority. For this purpose, the Atomic Energy Act and its complementary guidelines contain clearly defined rules and regulations. For beyond-design basis accident management measures, no such clearly defined rules exist. The actual implementation in the power plants is therefore often accompanied by considerable complications. This is due in particular to differing views on and interpretations of the regulations by the authorities but also among the utilities.

The different states of implementation of accident management procedures in German plants witnesses this facts. Although in the case of GKN II there is no fundamental disagreement between the authorities and the utility, the implementation of the accident management procedures has almost been completed. The actual carrying out of the modifications is sometimes rather difficult. A particularly forceful example highlights the existing problems:

A commercially available motor driven pump was selected as the ultimate feed capability in the secondary bleed and feed procedure. An integral part of this pump, to be stored in the emergency feedwater building, is a small petrol tank and a fuel store for petrol that at least a small amount of petrol would have to be stored in this petrol tank to facilitate a quick start when required. The applicable German regulation, however, permits "no flammable liquids in the emergency feedwater building" and the authorized expert appointed by the licensing authority was not willing to make any exceptions to accommodate the accident management procedures. After a dispute lasting almost one year an elaborate firewall enclosure was built at the pump location in order to finally make the rather small quantity of petrol in the pump's petrol tank tolerable. The price of the enclosure came to three times that of the pump and certainly does not facilitate a fast startup of the pump.

The above and other experiences of this kind have considerably subordinated initial enthusiasm over introducing additional accident management procedures. In addition there are substantial differences of opinion between the utilities and the licensing authorities concerning the verification of effectiveness of the proposed accident management measures and also on the depth of detail required for official approval. As these issues are particularly cost-sensitive, it is obvious that their clarification is of no small importance to the German utilities.
Furthermore, a recent court verdict (499) overruled the operation license of Obrigheim NPP invalid, causing additional confusion within the German utilities. One of the reasons for the decision was that, in the courts mind, the assessment of the implemented accident management provisions was not carried out in sufficient details before the license was issued.

Actually, the uncertainties how to handle licencing of accident management measures is the most severe problem for the utilities and can probably result in German stagnation for accident management implementations.

11. Training of Operating Personnel

The introduction of the new accident management procedures requires comprehensive training measures, especially for the control room personnel. The focusing on design basis accidents that was trained for many years had to be extended to the phenomena of beyond-design basis accidents. The initial considerable psychological barriers within the shift crews have now been overcome and accident management measures are a regular part of practical and theoretical training.

In co-operation with GKN II, the KSG/GRS Simulator Centre and GRS, the training of accident management procedures in a new pilot project on a full-scale plant simulator. In the framework of this project the simulation model of the training simulator was upgraded such that e.g. the thermohydraulic phenomena occurring during the simulation can be simulated with sufficient accuracy. In total three demonstration exercises were carried out with the upgraded simulator, the last one involving the crisis teams of both the owner and the vendor of the plant as well as GRS, who represented the general public and the media.

To determine the exact time span for manual actions some on site exercises were also performed during the project.

At the end some additional - unexpected insights came from the project:

- improvements in hardware configurations for easier and quicker handling of manual actions,

- improvements in the integration of training materials into the daily routine.
Today's point of view, not expected. By CRN, significant modifications of the existing accident management procedures are from particularly in the area of techniques of accident management processes. It will be supported development of accident management procedures through specific CRN research projects.

Even if they are clearly laid out and do not overtax the operating personnel, further events, the procedures developed possess great potential for further reducing the risk of core melt.

The procedures developed possess great potential for further reducing the risk of core melt are outstanding but under development procedure. Only provisions to cope with large scale hydrogen release into the containment CKN II has to a large extent implemented beyond design basis accident management

12. Conclusions and Outlook

Provisions such as secondary and primary bleed and feed procedures construction in France and UK are specified to cope with preventive accident management in addition to the new generation of simulators for CKN I and CKN II currently under development.

However, the responsibility for plan management in such cases will normally be taken over by the emergency response organization after about 1 hour. A sound knowledge of these phenomena is required in CKN II the use of all accident management procedures is not restricted for skill superintendents.

Due to the changes in the design of accident management procedures, the amount of activity released has become insensitive to the type of accident, as high levels of core damage have become more common. However, the increase in the amount of activity released is still a major concern for the future.

In the meantime, training of accident management personnel is still in progress, especially those responsible for prevention of core melt and less in-service.

- a complete restructuring of alarm management for the shift.
- optimization of the written procedures (reduced volume), and better training.
management measures within the German licensing procedure.

Authority of the licensing issue concerning the legal position of beyond-design accidents actually the main problem for German utilities is the classification by the licensing
<table>
<thead>
<tr>
<th>Abbreviation</th>
<th>Full Form</th>
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<tbody>
<tr>
<td>TVE</td>
<td>Three Mile Island</td>
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<tr>
<td>SBF</td>
<td>Secondary Bleed and Feed</td>
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<tr>
<td>RSK</td>
<td>Reactor Safety Commission</td>
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<td>PSA</td>
<td>Probabilistic Safety Analysis</td>
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<td>PB</td>
<td>Primary Bleed and Feed</td>
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<td>Loss of Coolant Accident</td>
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<td>GRS</td>
<td>Gesellschaft für Strahlen- und Umweltforschung (Limited Liability Company)</td>
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<td>EOP</td>
<td>Emergency Operating Procedure</td>
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<td>BMU</td>
<td>Bundesministerium für Umwelt</td>
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<td>BEM</td>
<td>Beyond-Degree-Basis Emergency Manual</td>
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<td>ATWS</td>
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Glossary
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Technical data

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### GKN II

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GKN II - Initiating Criteria for "Secondary-side bleed and feed"

- Criteria for preparation "Secondary-side bleed and feed"
  - 4 x 4 SG level < 4 m
    - Yes: Violation of critical safety function "SG-feed"
    - No: 3 of 4 SG level < 5 m
  - Or
    - No EDG available
      - Yes: Loss of offsite power
    - Yes: EDG available

EDG = Emergency Diesel Generator
### Core Cooling

- 2.1 No procedures as yet

### Subparagraphs

- 1.1 Emergency Measures

### Emergency Organisation

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<td>Water tank (passive)</td>
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In the emergency feeder building
The DZ supply is a special dedicated power supply

Filtered control room ventilation
Other Utilities

5.1
5.

Use of 380 V mobile diesel sets
Use of 20 kV grid
Restoration of DZ-supply

4.3.2
4.3.1
4.3

Station blackout
Restore 110 kV energy supply after

4.2

Electric Energy Supply
6 km away

4.1.2
4.1.1
4.1

From 120 MW gas turbine (located from unit I

Electric energy supply

4.

Supply of containment atmosphere
Containment venting
Containment integrity

3.2.2
3.2.1
3.2

Modification of SC safety valve setpoint
Containment isolation
Limitation of releases to environment
Radiation retention

3.1.2
3.1.1
3.1
3.

Emergency Measures (Page 2)
Problems of Probabilistic Safety Assessment (PSA) and development of severe accident management have been actively tackled with the study of severe accidents including both so-called severe accidents. The paper describes briefly the accident beyond the design basis, namely at the TMI accident in the USA in 1979, as well as the Japanese accident in Umeå in 1996. By the latter, the frequency of reactor trips is much lower than the Japanese community. However, the nuclear power plants in Japan are kept in accident safety, which is demonstrated.

2. Introduction

This paper describes briefly the accident management strategies for PWR plants in Japan.

1. Abstract

I. Presented at the OECD Workshop on Severe Accident Management Implementation

II. From 12th-14th June 1995

III. Nishic, Conference USA

The Kansas Electric Power Co., Inc.
General Office of Nuclear and Fossil Power Production
K. Shigemune, K. Yoshikawa and M. Ohnari

for PWR Plants in Japan

Severe Accident Management Strategies
On the other hand, the results of the Accident Management Exercise are based on the methodology described in IAEA Basic Safety Standards, and are intended to evaluate the Accident Management Exercise and identify areas for improvement. The results of the Accident Management Exercise are compared with the criteria outlined in the IAEA Basic Safety Standards to determine if the NPP is operating in a safe and effective manner.

In the event of a severe accident, the NPP Safety Team is responsible for implementing emergency procedures and coordinating response efforts. This includes activating the on-site Emergency Operation Center (EOC) and coordinating with local and regional authorities. The EOC is staffed with trained personnel who are responsible for coordinating the activities of the various emergency response teams.

The NPP Safety Team also provides regular updates to the public and other stakeholders on the status of the emergency, including information on the extent of the damage and the steps being taken to mitigate the situation. This information is disseminated through a variety of channels, including social media, local news outlets, and public briefings.

In conclusion, the results of the Accident Management Exercise are an important tool for evaluating the effectiveness of a NPP's Accident Management Plan. These results are used to identify areas for improvement and ensure that the NPP is prepared to respond effectively to severe accidents.
In the case of ECCS recirculation failure during LOCA, information mode can be continued by
continuous injection of make-up of water through SGR.

Because it has sufficient steam dumping capacity for cooling the RCS through SGR, the RCS is expected to be at
charged continuously under the SGR system, which is linked by the safety injection (SI) system. However, they are normally in the system, with carryover heat from the RCS to the secondary system, can be cooled by steam generator (SG), with carryover heat from the RCS to the secondary system, to be in action for cooling the core in case of HPS failure. The reactor coolant system (RCS) during a medium or small LOCA, the high pressure injection system (HPIS) is expected to be

Utilization of turbine bypass system

Because power of the NPPs pumps are supplied through the non-functional buses.

If the bypass mode of the turbine bypass system is not available, the turbine bypass system, which is usually connected into the feedwater system, can remove unwanted heat from the core by the secondary system.

(1) Deviation of core cooling by the secondary system

- Figure 3. Each scheme is connected described below.
- Safety functions were considered. Accident management strategies are shown in Figure 2, and management strategies for each scheme's accident scenarios. Accident management strategies were developed as accident management strategies. The counter measures against the cases which the occurred after event accident, prevention severe core damage and/or CV failure, by also cases which this resulted have to be considered. These safety functions are necessary not only safety functions required for accident management strategies and their effectiveness.

- Accident management strategies and their effectiveness can be prepared due to the relatively small CV volume and low CV design pressure.
supplemental CV iso-pressures, achieving easier heat removal by natural convection. Also, if core-concrete interaction occurs, CV pressure was sufficiently suppressed by components performance, another advantage. Tests were made with SWAP code, the cooling coil can significantly remove core decay heat. To confirm the CV cooling under normal convection or supplemented with pressurized water in the event of a transient or core damage scenario, the test removed the heat from melts of cooling coil and ventilation system as well. In case of unsuccessful component cooling, such as failure of component spray system, component cooling by natural convection If was confirmed that this series of operations can be carried out in an appropriate way.

PHRS in the long term cooling in addition, recirculation by feed-and-bleed with HPS and PORV can replace for In recirculation, recirculation pressure (PORV), pressure (PORV) has been considered for the HPS system, which can substitute for the recirculation pressure (PORV) and main/primary pressurized water system (PWR). In this strategy, non-safeguarded, in the residual heat removal system (PHRS), 

residual SGs be depressurized by the HPIS, meanwhile the RCS will be cooled down by the RCS should be depressurized by the HPIS, meanwhile the RCS will be cooled down by the RCS...
1. It is confirmed that both of the alternate measures have sufficient cooling capacity.

2. Alternate water resources such as the new water tanks or another cooling water system for returning time of CWS recovery. Component cooling water can be supplied from the system, accessible in a situation that the component cooling water system is in accident management emergency, but also in normal operation because the minor pumps should be cooled by CWS.

3. Component cooling water system (CWS) is one of the important systems not only in emergency but also in normal operation because the minor pumps should be cooled by CWS.

4. Alternative auxiliary component cooling.

5. The experiment showed that the integrity of CWS can be maintained by controlled pumping with inhibitors. To obtain the pressure suppression effect another MAPH run, the pressure was reduced to 8%.

6. A MAPH code analysis was made for the 4-loop plant with ice condenser C/W. The full pressure with inhibitors.

7. During a severe accident, hydrogen could be generated over design basis. This measure

8. (applied only for the plants with ice condenser C/W)

9. Controlled pumping of hydrogen

10. The capacity of PORV was confirmed to be sufficient to decrease the RCS pressure.

11. Depressurized with the PORV in order to prevent direct containment breach.

12. In case that 1-2 vessel maintains pressure in high pressure condition, the RCS can be forced depressurization of the RCS.
In conclusion, we have developed accident management strategies based on the knowledge on the severe accident and plant characteristics obtained from PSA. The Japanese PWRs are operated in satisfactory safety level from the viewpoint of core damage frequency and CV failure probability. Regarding CV failure, there is noticeable difference depending on CV type, in other words, on either with or without ice-condenser.

In this paper, we also explained that 10 strategies which were identified for core damage prevention and CV integrity.

Effectiveness of the strategies which needs numerical analysis in detail is verified by the computer codes, some of whose results are referred in this paper.

We intend to prepare plant-specific procedures and modify some facilities based on these accident management strategies. At the same time, we will improve the existing implementation system and education/training method concerning accident management for further safety enhancement.

We believe that the strategies will bring about confidence in nuclear safety among general public and further understanding and acceptance of nuclear power generation.

Acknowledgements

The authors are grateful to other Japanese PWR utilities and the Japanese PWG, NISSEI vendor for their cooperation on developing accident management strategies.

References

2. 1989 ASME/ASCE Annual Meeting, "Tea and Mass Transfer around One-Row Tube Bank Simulating Containment Fun Cooler under Post-Accident Condition in Nuclear Power Plant"
Figure 2: Level-2 PSA Results
<table>
<thead>
<tr>
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<td>Safety Function Support to</td>
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<td>- Control Bumping of Hydrogen - Forced Depressurization of the RCS - Water Injection into CV - Containment Cooling by Natural Convection</td>
<td>Reactivity Containment of</td>
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<td>- Diversification of Core Cooling by Secondary System</td>
<td>Reactor Shutdown</td>
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</table>

Table-2 Accident Management Strategies

Table-1 Safety Feature of Typical Plants
Figure 4: Core Cooling Performance

Time (hr)

Fuel Temperature (°C)

Core Water Level (m)

Figure 3: Typical AM Strategies and Facilities in PWR
Figure 5: CV Cooling Performance

Figure 6: CV Cooling Performance

Start of water feed to cooling coil

CM Pressure analysis by MAP code
Figure 8: MAPHY-BURN Result

(CY Pressure)

Pressure (kg/cm²) vs. Time (hr)
June 14th, 1995
SESM Conference on Severe Accident Management
Presented at the

Martin L. Van Halem

By

Northeast Utilities

An

Severe Accident Management
In order to make this effort a successful one, we will rely on the experience and knowledge of our operators and technical support staff in order to make this effort a successful one.

This paper will describe some of the aspects of NRC's organization and structure which pose unique challenges in implementing SARM. Our approach to meeting these challenges builds on the experience which we have gained in Emergency Plan Organizations and Exercises. We have developed guidelines in Implementing SARM. Our approach to meeting these challenges builds on the experience which we have gained in Emergency Plan Organizations and Exercises.

The Guidelines and Strategies:

SARM Self-Evaluation:
- Definition of decision making responsibilities
- Information (instrumentation) needs assessment
- Communication aids
- Training
- The guidelines and strategies

The Elements of Severe Accident Management Implementation identified in NEL 91-04 are:

Establish a means to consider and possibly adopt new severe accident information

Incorporate the severe accident material into appropriate training programs

Integrate the Severe Accident Management Guidelines with the Emergency Plan

Specific Guidance:

Independent Plan Examinations performed for our units and implementation plan

Review the industry developed guidelines and technical basis and the results of the

As described in 19-04 Rev. 1, we have committed to complete our implementation by December 1998. Implementation of Severe Accident Management as described in the NEL formal industry position 91-04 Rev. 1. We have committed to complete

Northeast Utilities like all US utilities is committed to implementing Severe Accident Management
applied approach uses selected instrumentation to determine whether any of the SWM functions are not the concept of critical severe accident management functions and challenges. The WAC

The WASMeagle Owners Group has built its severe accident management guidelines around different SAMG's at the WASMeagle site poses a significant challenge.
implementation issues are fairly straightforward. For NU, however, the issue of integrating these
Response Organization (especially TSC) are only responsible for the severe type, the
all participants. For Utilities which have only one NSSS type at a site and whose Emergency
accident, defining the knowledge and skill levels of the TSC and providing appropriate training for
involve defining the roles and responsibilities of the operations staff and the TSC during a severe
utility specific implementation issues must be addressed. The key issues in implementation
utility between the SAMG and the Emergency Response Organization. This is where many of the
issues between the Operations staff and the TSC. The other important issue for the SAMG is
consistent with their Emergency Response Guidelines; each of the Owners' Groups have established
groups have also developed guidelines for use by the control room. By developing the SAMG's
The SAMG's are generally intended for use by the Technical Support Center. Some Owners'
responsibilities of the TSC and the Operations staff change significantly among the SAMG's.
result, the decisions made are reflective, lessons learned, and to some extent, the roles and
form and philosophy of their Response Emergency Response Guidelines (ERG) in mind. As a
were developed with the
developed severe Accident Management Guidelines. These SAMG's were developed with the
implementation severe Accident Management. Each of the US NSSS vendors Owners' Groups have

The diversity of utility which NU is responsible for poses a particularly difficult challenge in
and Organization. This paper will address the implementation of SWM under that plan.

Millstone Section A(1), BWIR II (1), MB-2 (1, CE, PWR), and MB-3 (1, CE, PWR), and through
Northern Utilities currently operates five Nuclear Power Plants at three locations: Haddam Neck

MWRI, 1960's vintage W PWR), Sequoyah Section (1 W PWR PWR) and through

9, 10, and 11 loops (a 1960's vintage W PWR), and through

9, 10, and 11 loops (a 1960's vintage W PWR).
Severe Accidental Conditions

Support guidelines to assist them in optimizing the SAG and EPC sectional control curves for some of the SAG to the TSC of the Control Room. In addition, the TSC will have technical specialists as well as the specialties for defining with H2 in contamination. The utility may decide to assign accidental guidelines from the EPG. The SAG consists of the integrated containment bytwinning direction in the EPG. They have since modified the approach and now have separated severe accident management reviews of the BWR Owners Group initially choose to include all of their severe accident management.

The CEOSevere Accident Management Guidelines are built around the concept of Plan exchange once the TSC is operational.

Guideline documents are provided for both the TSC and the Control Room. The control room guidelines provide section A.1 and facilitate the information of the detailed discussions. Potential mitigation strategies are presented for each function which is not being adequately met. A WDG (waste discharge) is directed toward mitigating that challenge. If no severe challenge is identified then the WDG guidelines lead the evaluation through a prioritized evaluation of each SAG function. If there is a severe challenge in the containment, it presents then.
Operations staff have a complete knowledge of the physical locations and capabilities of the plan. TSC resources needed to mitigate the ever-present hazmat situation in the TSC and the SAWG. However, both centers (Control Room and Hazard Operation Teams) are essential to the TSC and the SAWG. The phenomenon and behavior which can occur during a severe accident involves additional responsibilities for the Control Room staff. The TSC is not responsible at this point, and responsibilities are shared.

When an event progresses to the point where phenomena are occurring which were not considered in the development of the emergency, the control room is considered to be self-sufficient. The Self-Support Center is a concept with which many people dealing with operations and TSC staff are not familiar. In the Self-Support Center, many documents speak about Control being transferred to a point or the regulatory accident when an accident occurs. What is the role of the Control Room during a severe accident? Is it accountable for "Control," or is it a transfer of authority? How can we integrate the TSC into this situation when the operators and the 'labeled' personnel? Which is responsible for "Control," to whom should be transferred, and what skills are required at each point?
In SHM and to participate in the evaluation process, what level of training is required of the operations staff to familiarize them of the role of the TSC?

Accident Management Guideline:
Designated AM Team members will have a general familiarity with each plan and several designated AM Team members will be selected from the list of experts for that unit. All of the events occur at a plane which is not one of the two that the call AMT has specialized on then team will be required to be an expert on the plan systems and SWMs for two of the plans. If an event occurs within a plane which is not aligned to one of the call AMT or call-in members of the TSC, it is required to be an expert on the two that the call AMT has specialized on then team will be required to be an expert on the plan systems and SWMs for two of the plans. If an event occurs within a plane which is not aligned to one of the call AMT or call-in members of the TSC, it is required to be an expert on the two that the call AMT has specialized on then team will be required to be an expert on the plan systems and SWMs for two of the plans.

In NTS, the Emergency Plan is Accident Management Team are so-called common responders. It is

SAMG's. This role is consistent with the Evaluate position defined in NE1 94.

The TSC is comprised of engineers who are familiar with accident analysis, PRA, and

Group's SAMG's be implemented without requiring extensive training?

If common responders are used for all units how can the diversion approaches used in the Qmers

What are the knowledge and skills required for the TSC staff responsible for the SAM guidelines?

In the search for solutions to the severe accident contamination exercise identified the importance of keeping the control room staff actively involved by given training in the severe accident environment. Both the CEC and the WOC alike have contributed significantly in the response of the AMT staff to be evaluating the pros and cons of the TSC is performing and should be working with the TSC staff to identify and implement the severe accident. The control room staff should be constantly aware of the assessments which mind we believe that the control room and the TSC must both be actively involved in responding to
The development of the new ERO plan coincides with the implementation of a comprehensive human factors program. This program includes the following key areas:

1. **Safety Culture**: Creating a culture where safety is at the forefront of all activities.
2. **Human Factors Training**: Regular training to understand human error and how to mitigate it.
3. **Communication and Collaboration**: Enhancing communication between departments to prevent errors.

These initiatives are designed to reduce the risk of accidents and improve overall safety in the workplace. The implementation of these programs will be monitored and evaluated regularly to assess their effectiveness.

Furthermore, the case study illustrates how a proactive approach to safety can lead to significant improvements in workplace safety. By addressing human factors, organizations can create a safer environment for all employees.

In summary, the new ERO plan and the human factors program are integral parts of the organization's strategy to enhance safety and prevent accidents. The continued implementation and evaluation of these programs will be crucial in achieving these goals.
References

work at Operations is a full participant in the SAW process.

Despite the TSC in order to reduce the burden on the Operations, we must remember that SAW will only

struggle to enhance our capabilities. Even though the SAWs were written primarily for use by

use: We must use those tools carefully by modifying our current practice to take advantage of their

responsibility. The Guidelines developed by the Industry are in a sense a new set of tools for us to

1. NEI 91-04, Revision 1, "Severe Accident Issue Closeout Guidelines," December

2. CE-QSD-916, "Generic Accident Management Guidelines, CEC Task 726,


4. BWRO Owners Group "Accident Management Guidelines Overview Document,

5. Revision 0.5, January, 1995

6. WCAP-14213, "Westinghouse Owners Group Severe Accident Management

Guidance Valuation," October, 1994
Set up Training

Define Performance Requirements

Issue and Control SAM Documents

Define Roles and Responsibilities

Integrate with Emergency Response Organization

Test with Table Top Drills

Write Specific Procedures

Define Links to TSC

Provide Control Room Documents

Modify EOP’s as Required

Choose Transition to SAMS

Integrate with Operations Documents and Training

Procedures

Identify All Items Equipment use and limits, Pre-Written

Identify equipment capabilities

Calculate Plan specific responsibilities, create calculation Alids and

Compensate for Plan specific differences

Addition of Plan specific capabilities

Convert to Plant specific

Simple Table Top exercises

Screen against IPE & IPEE results

Identify non-applicable instrumentation, strategies

Conversion of SAMS to Plant Specific
Update with new severe accident information

Re-Training Requirements and Lesson Plans

Final Confirmation Exercises

Confirmation Training and Use of SAMG

Accident Management Team

ERO Decision maker (ADTSC)

Operations / STA
which were submitted to MITI in March 1994.

plan and worked out AM strategies based on worldwide SA research and PSA results. PSAAS for individual plans. Electric power companies carried out PSA for individual plans. As electric power companies reported their plans for AM strategies with international trade and industry, which is a regulatory body for nuclear power strategies voluntarily. To respond to the recommendation, MITI(Ministry of recommended nuclear power plans. Licenses to implement continuous and effective AM strategies for SA in nuclear power plants. Are SA have been conducted worldwide. These knowledge enables us to investigate since the accident at Three Mile Island (TMI-2), SA research has been accelerated and would dominate the risks of NPPs.

in 1975, the report indicated that multiple failure of safety systems could lead to SA. The risks of NPPs were evaluated for the first time in the report. WASH-1400 issued 2. Introduction

Introduction.

Technical basis. And finally on going activities for AM implementation are shown. Describes typical AM strategies selected based on PSA with their effect and configuration Japanese BWR, configuration with their PSA results, and regulation, utilization, spontaneous activities.

This paper introduces Japanese BWR's, configuration with their PSA results, and regulation, utilization, spontaneous activities.

This high safety level of Japanese BWR NPPs have been assured by high quality of Japanese BWR Nuclear Power Plant's (NPPs) are recognized to be in high safety level. The results of PSA satisfy the IAEA INSAG's safety standard very well and individual plants.

AM strategies we conducted and referred Probabilistic Safety Analysis(PSA) for implementation of AM for verification as well as prevention. In the process of selecting prevention, however, Japanese BWR Utilities are voluntarily participating further Accident Management(AM) has been developed already for severe accident(Severe Accident Management(AM))

1. Abstract

Chiyoda-Ku, Tokyo 100 Japan
Tokyo Electric Power Company, J. Uchida, M. I. Kawai, K. Kamo, S. Nomoto
Yamato, 13, Minami-Ku, Tokyo 157 Japan

Accident Management for BWR in Japan
Figure 2. Containment types of Japanese BWR plants

Table 2. Water injection & decay heat removal system

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Table 1. Types of Japanese NPPs and their numbers

Table 1 shows the types of Japanese BWR NPPs and their numbers. These types are classified into four plans have their own type of containment such as Mark-1, Mark-2, and Mark-3. The PSA is mostly affected by the reactor type, on the other hand, the containment type has large influence on Level 2 PSA. Table 1 shows the types of Japanese BWR NPPs and their numbers. These types of Japanese NPPs are so standardized that they can be categorized into four plans.
**Figure 2** shows the Level-1 PSA results for four reference plants of each reactor type.

The process of evaluating only component failure frequency without source term unavailability of mitigation features. In this paper, we use the term "Level-2 PSA" as Level-2 PSA were calculated by referring to PSA research and its analysis with some year. Success criteria and branch probabilities used for the probabilistic event tree of each assumed conservative one: event occurrence during accidental operating reactor initial event occurred at the plant. This means that the event was mostly calculated. For example, in the case where there are two mean repair time were mostly US data, in the case that these began, the other hand, equipment failure data and other accumulated data such as experience. The PSA conducted this time are for internal events only. The initiating event...
The results of Level-2 PSA are shown in Figure 3.

Power supply system

CDF and BDF are low. As a result, SBO sequence rises to the surface but still is frequent. The measures were selected targeting Level-1 PSA. So the recognitions of major sequences in the design phase of ABWR, combination of safety system with other SA counter

Reliability for high pressure core cooling.

In the design phase of ABWR, combination of safety system with other SA counter

Reliability for high pressure core cooling.

Isolation condenser (IC) which is well known passive system and has high

Fewer ECCS pumps compared with BWRA 5 and ABWR. On the other hand, BWRA 3 has

BWRA 3 has relatively large contribution of TLOCA and negligible small contribution

TLOCA: Transient followed by failure of decay heat removal
TOUX: Failure of high pressure water makeup and depressurization
SB0: Station Blackout
AWMS: Auxiliary Mains Lost without Steam
TMOV: Transient followed by failure of water makeup
LOCA: Loss of Coolant Accident

The meaning of abbreviations in the figure is listed below.
- Support function for safety system
- Heat removal function from containment
- Water injection function into core and/or containment
- Reactor shutdown function

Effective to present and mitigate SA consequences. These functions are as follows:

Conditioning PSA, we learned that enhancement of the safety functions are
depressurization during high pressure injection system failure and so on. By
existing AM strategies are manual SC injection during Reactivity accident, manual
some modification of component and system were necessary.

To the modification of PSA, we have developed operational procedures for preventing
SA without any component change. After the recommendation, we came to pay attention
on the prevention of SA and we have developed operational procedures for preventing
before NSC's recommendation. Without main efforts on AM activities have focused

4. AM strategies as countermeasures for SA

 containment failure mode.

This sequence makes steam explosion the second top contributor for Mark-III.
In the Mark-III containment, the motion fuel is cooled down.

In the Mark-I containment, the motion fuel, swelling through the pedestal,
the containment shell will break and some contribution in the plant with Mark-I.
In the Mark-II containment, once the motion fuel was swelled into the pedestal area,
moist in the SBO sequence and main contributor in CD for ABWR is SBO.

Containment failure of RBWR with Mark-II containment. The largest contributor for
combined with Mark-I containment. On the other hand, the DOR contributes to the

The OT mode of containment failure is the main contributor for RBWR, which are

\[ \text{SE: Steam Explosion at well-wet} \]
\[ \text{OF: Over Pressure due to loss of decay heat removal} \]
\[ \text{DCR: Direct Containment Heating} \]
\[ \text{OPA: Over Pressure after ATWS} \]
\[ \text{SAP: Containment Shell Aid} \]
\[ \text{OL: Leakase due to Over Temperature} \]

The meaning of abbreviations in the figure is listed below.
Table 3b Selected Mitigation Strategies

<table>
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Table 3b Selected Mitigation Strategies

Table 3a Selected Mitigation Strategies

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Table 3a and 3b show the AM strategies with their corresponding function and SA sequences.
Injected into containment if can cool containment atmosphere and molten fuel. Containment, when the water is injected into RVP it can cool core fuel. When water is not optimally designed for injecting water into Reactor Pressure Vessel (RPV) not worst case strategy utilizing make-up water system and the protection system which arc over containment and shell glitch modes. The system is shown in Figure 5.

Alternate water injection into the containment is selected against Water injection function - Figure 4 AR1 system.
Confinement over pressure is so slow phenomenon that residual heat removal can delay further confinement depressurization. If confinement pressure are recovered before confinement failure, and alternative confinement over pressure is.

Figure 6: Alternative heat removal

- Heat removal function

Selected sequences TV sequence.

Figure 6 shows alternative heat removal and hardened wall well venting which are.

(1) Signal.

Useful. This type of automatic depressurization is initiated only by low water level.

Depressurization has some uncertainty: automatic depressurization for transient become manually in order to accelerate low pressure function. Because manual addition another initiation signal to the ADS initiation signals, which is selected sequence TOX sequence, is.

Emergency Core Cooling System (ECCS) in the LORA event, simultaneous signals of Automatic Depressurization System (ADPS) exists in the current BWR as one of the preventing motion fuel spreading and core-concrete interaction.
The effect of these AM strategies are discussed in the next chapter.

Because AM strategies are adopted on each Reactor type based on individual PSA.

![Diagram](image)

**Figure 7:** Electrical accommodation

---

The other hand, low voltage emergency power utilization needs some modification of equipment.

High voltage one makes use of common bus to supply excess emergency power. On

against SO subcases and DCH mode. This AM strategies are shown in Figure 7. Low and high voltage emergency power utilization from adjacent unit are selected.

- Support function

-atmosphere, accompanied FP across Reduced drastically.

steam in the containment is scrubbed through well-wet well water before being released into

can also mitigate over pressure mode in an accident even after core damage. Because, if

reduce containment pressure. This AM strategy can prevent core damage, however, it
Injection into RPV. The debris temperature could be suppressed under 1200\degree C.

Figure 9 shows the NPD analysis for debris bunch using alternative water attack. Debris coolability and containment atmospheric cooling are discussed about the effects of SA mitigation in the containment with regard to shell thickness. A strategy can prevent core damage if water is injected into RPV timely. Here alternative water injection.

Figure 8. Effect of RPT and AR1

Containment over pressure enough to prevent core from degrading after occurrence of ATWS and accompanied AR1 can shutdown the plant within a few minutes. This AR strategy is effective as shown in Figure 8, RPT can reduce reactor power without over pressure of RPV.

- Alternative Reactivity Control

Research and analysis is discussed here.

Selecting AR strategies, the effect of AR strategies with their corresponding SA heating, core-concrete interaction, aerosol pool scrubbing have been investigated in various phenomena in the containment such as steam explosion, direct containment.

5. Effect of AR strategies
strategies is to vent steam from wet-well through pool water when steam goes through
from containment to atmosphere and suppress the containment pressure. Selected AM
The hardened wet-well venting is a kind of pressure relief valve that releases steam
removal efficiency becomes high. When the dry-well containment is high and the reactor pressure is high, the heat
containment pressure might be suppressed or delayed to increase from states to states.
The capability of alternative heat removal depends on the plant status and the

- Alternative heat removal and hardened wet-well venting

- Automatic depressurization

We expect another study on this phenomenon will be conducted in the future.
MAP analysis shows that the injected water can cool the debris at ex-vessel stage.
Motion debris showed the possibility that shell attack might be prevented by water injection onto
was developed based on the study and the analysis of an actual Mark-1 containment
containing whether the water lying on the dry-well floor can prevent debris from

Japanese BWR units and NSSS vendors conducted experimental study which

Figure 9 - Effectiveness of alternative water injection
FIGURE 10. Accident management organization and guidelines

- Figure 10 shows the relation between operational procedures and emergency management.

Existing resources can be used with a little or no modification for accident accident and provide comprehensive accident management aids system. Basically the development of operational procedures and guides, enhance institutional assessment severe.

In order to fully implement accident management, we need to arrange organization.
- On going AM activities.

These AM strategies can reduce the plant's residual risk effectively.

- Plants are failed from common cause failure.
- Diesel generators and to operate switch gears in high voltage line it all batches in one operational guidelines. Low voltage station allows operators to initialize emergency procedures on sharing high voltage emergency electrical power has already been be safe shutdown states even when SBO sequence occurs.

Since Japanese NPPs have multi-plant, electricity power supply can be easily
- Emergency power utilization from adjacent unit.

Japanese BWR units and NSSS vendors. (12)
be completed around the beginning of the next century. Detailed system designs are being investigated. So the full AM measures are expected to be implemented. By the AM strategies, the safety margin of NPPs will be increased by these AM strategies. AM strategies are selected based on IES and the effect of the AM strategies are NPS have been assessed also in light of the PSY's point of view.

7. Conclusion

Establish more effective accident management and on these guidelines and other resources to be arranged, we are going to develop against possible accidents. To identify and categorize operational procedures which include concrete operational measures of system and accident. In addition to the accident management guidelines, control room operators would need measures, and identification of system operators would be required in order to maintain awareness of safe conditions. To this end, the guidelines will be developed.
OECD SPECIALIST MEETING ON
SEVERE ACCIDENT MANAGEMENT IMPLEMENTATION
Niantic, Connecticut, USA; 12th-14th June 1995

Report of the
Nuclear Power Station Gundremmingen (KRB II)
Germany
Severe Accident Management

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Abstract

The indispensable precondition to cope with events and accidents, no matter whether they are within the design range or beyond, is a good designed plant and a highly motivated and trained crew. The following point turned out to be most important to cope with severe accidents: A sophisticated and ergonomic approach and procedure supporting the personnel in order to make the decisions for the best appropriate action at the given time.

In the early eighties we developed a symptom-oriented (state-oriented) procedure for the decision making of the response actions within the design range ("Emergency Operating Procedure", EOP). At that time we were the first utility in Germany applying such an approach. After the terrible accident of Chernobyl in 1986 we created "Emergency Actions" (EA) (another usual word is "Accident Management Measures") and extended this procedure to the range beyond the design to cope also as far as possible with severe accidents.

It is important to say, that our EOP comprises, in the same paper, all decision making for actions in design and for Emergency Actions beyond design as well. In the terminology of the IAEA, Vienna, our EOP covers the EOP, the AMP (Accident Management Procedures) and The AMG (Accident Management Guidelines). In our experience it turned out very advantageous to have only one paper for all affected personnel e.g. the shift crew and the Emergency Response Organisation (ERO) (Crisis Team).

We elaborated very carefully more than 90 Emergency Actions (EA) for each of our two units. About half of them are to recover the power supply on the Diesel busbars. The enormous number of actions on this field is due to the fact, that we prepared individual descriptions for each possible busbar / Diesel connection between our 10 bars at the station. As far as possible we tested the performance of each EA, the time needed to bring them into function and how it works. In cases when no test is possible in reality, we made thorough evaluations as to what performance is achievable.

Since we got our new excellent fullscope simulator in 1993 we immediately tested how our approach and EOP works. We made tests with 22 events increasing all gradually to the range beyond design. We also tried to find out the best appropriate casting among the shift crew to apply the EOP. In sequence we trained all shift crews with the application of the EOP. Essential findings were that the approach works in an outstanding manner. Beside there were no principle changes on the EOP necessary, we figured out that a lot of improvements have been possible. Further it turned out that the instrumentation needed for the application of the EOP has to be marked and set off out of the lot of the other instruments and that other technical improvements in the plant would be of advantage also.

We elaborated a training program for all personnel involved in carrying out the EA itself. This training includes beside theoretical explanations real processing of the EA as far as possible without affecting the normal operation of the plant. In other cases a mock-up training is being performed.
In our effort to cope with severe accidents we also have to deal with several uncertainties for example avoiding a steam explosion in the containment, recriticality during reflooding of the core, hydrogen problems, instrumentation problems, common mode failure and so on.

1. **Strategy to cope with events and accidents**

The safety of nuclear power plants is guaranteed in the plant design by a number of quality assurance measures during planning, construction, erection and operation. Keywords in this connection are „basic safety“, „redundancy“, „diversity“, „service-tested components“, „reliable working automatic systems“ ¹), „qualified personnel“.

A survey about the general strategy to prevent and mitigate accidents within and beyond design is shown in Fig. 1/1.

Probabilistic system analyses (PSAs) yield a residual probability of $10^{-6}$ to $10^{-5}$ cases per reactor and year for a core meltdown accident to occur (Fig. 1/2). This „residual risk“ can be considerably reduced by „Emergency Actions“. Another decisive factor for the introduction of Emergency Actions was the fact that they can mitigate the development and the effects of accidents which have to be assumed nevertheless to such an extent that the environment generally remains unaffected.

These plant-internal Emergency Actions (Am. English: „Accident Management Measures“ serve to set up another Safety level which surpasses the design-basis safety-system. This additional safety level 4 fits well into the „defence-in-depth“ concept applied so far (see the issues of the International Nuclear Safety Advisory Group, INSAG or the papers of the IAEA, Vienna).

Emergency Actions include the mobilisation of reserves in existing systems, special operating mode, linking of normally separated systems and connection of facilities established especially for this purpose. Emergency Actions are almost exclusively manual actions.

94 individual Emergency Actions have been prepared for every unit of the Gundremmingen nuclear power plant (KRB II). Of these Emergency Actions, 54 alone are solely for power supply restoration (see section 4).

The implementing instructions for Emergency Actions are set down in our „Emergency Manual“. This manual comprises of four files (see section 3).

¹ (In order to maintain the reliability of these automatic systems we installed a computer aided system that checks the proper function when ever an automatic system was launched irrespective if due to a real demand or of an in service inspection . The results of this post trip review system „NOVA“ are available to the operators a few minutes after the automatic systems have started. It turned out as very helpful and as an effectful step to enhance the safety of our plant (separate paper available).
Great importance is attributed to the training of the personnel making the decisions to initiate and handling these Emergency Actions. This is the only way to establish the necessary degree of confidence in the actions taken (see section 5).

It turned out that the most important part to cope with any accident is a sophisticated and ergonomic approach and procedure supporting the personnel in order to make the decision for the best appropriate action at the given time.

In our case the criteria for the initiation of all actions, irrespective whether they are operational actions, safety actions or Emergency Actions are laid down in the „Emergency Operating Procedure (EOP” (see section 2).

To support the shift supervisor in his tasks during an accident, for linking the conducts to outside organization and institutions and for important decisions regarding release of radioactivity, flooding of parts of the containment etc. we installed an effective Emergency Response Organization (ERO, Fig. 1/3), conducted by the normal operating management but direct and tight organized.

In the past we took two practices to figure out if there are any weak points in this organization. We intend to go on with this custom in order to improve the co-operation among the various concerned parts. We also consider to include our full-scope simulator in this practice.

2. Emergency Operating Procedures

2.1 Introduction

A symptom-based instruction procedure, the so-called "Emergency Operating Procedure EOP" was introduced for the management of accidents in accordance with the proposals of the Nuclear Safety Analysis Centre right at the beginning when the two boiling-water reactor units KRB II (2 x 1310 MW) where commissioned in 1984. At that time, we were the first utility in Germany using this approach. The application of this procedure proved to be successful. Following the accidents of Harrisburg and Chernobyl, extensive Emergency Actions were developed and prepared for the management of beyond-design accidents. The criteria set for the initiation of these Emergency Actions were also included in this EOP.

With the EOP, actions are inferred from the transient state of the plant. This procedure is

- quick,
- corresponds to the natural approach taken by the operator and
- covers all conceivable accident sequences.

With this procedure, the type of accident and the cause of the incident do not matter.
The operator has to observe the development of certain safety parameters and infers from these and other process variables his manual actions.

These are interventions

- to reduce the ongoing transients,
- to control the triggered automatic protective action,
- to initiate Emergency Actions and
- to bring the plant to a safe final condition.

The EOP is used for all accidents, no matter whether in design or beyond design.

2.2 How we proceed in case of an incident or accident and explanation of the EOP

Before we are going to start into detailed description of the EOP you should get an overview how we proceed in Gundremmingen in the case of an incident or accident (Fig. 2/1).

If there is any incident, indicated by alarms or other announcements, the shift crew has to prove if there is a deviation from the normal operation. Usually this job will be done by the Reactor Protecting System. If the answer is positive the reactor has to be scrammed.

For the first minutes after scram, we have prepared a so called "Post Trip Check". It is helpful to calm the nerves of the shift crew and ensures to get a quick overview of the plant status in a well organised manner. In this step no operator actions are required yet. It is only to get the necessary information about the plant situation.

The Post Trip Check comprises three pages to conduct by the

- reactor operator
- auxiliary systems operator and the
- deputy shift supervisor.

We like to show you the example of the operators page (Fig. 2/2).

In the last column the cases are indicated in which the operator should report the matter to the other members of the crew.

When the Post Trip Check is done, the shift personnel has to process the symptom-based EOP. This is a must, not a proposal!

Only in very unequivocal situations, when a typical incident is identified without doubt, the operators can additionally use the event oriented approach. We have concise and brief event-oriented procedures for all design basis accidents.
The step by step programmes to conduct the required actions for design basis accidents are in the Operating Manual. The programmes for the Emergency Actions are prepared in the Emergency Manual. The decision making for Emergency Actions will be made in the EOP without any exception.

But let us have a look in the EOP:

It's a fact that the more serious the events are growing, the more complicated the appropriate actions are to be figured out and to be performed. This correlation is shown in Fig. 1/2. Due to this fact we tried to develop and to design an EOP as simple as possible. Other objectives for the design of the EOP have been:

The EOP has to be
- as comprehensive as necessary
- as clearly arranged as possible
- unequivocally
- manageable
- well tested (plant, simulator).

We hope we actually were successful in the attempt to fulfil all these objectives. Let us see the example:

The EOP works in levels:

Level 1:

We start in processing the EOP with either the triggering of a scram or the loss of power on a Diesel busbar (Fig. 2/4).

In level one the entry to the action modules is shown. The first column shows the Safety Objectives.

The second column contains the Safety Parameters. The principle of the EOP is that if these Safety Parameters are kept in their normal range or - depending on the current situation - in an adequate range, the Safety Objectives are maintained. The third column shows the limits defined for the access to an Action Module. The last column shows the Action Modules.

If for example the RPV level is lower than 13.91 m, we have to enter the Action Module „RPV Level Control F“. The EOP is arranged in registers. For each module we have a separate register. So we have to look at register „F“: (Fig. 2/5)
Level 2:

Every first page of an Action Module contains a query logic. As can be seen, the query logic leads from certain limits on the Safety Parameter and the query of other parameters, system statuses and so on, to a certain action group. This action group is marked by a grading symbol and a characteristic numeral. There is also given a brief clear text what has to be done in this action group. So this level 1 comprises most of the normally needed information.

Let us make an example how this level works:
- RPV level < 12.35 m &
- feedwater lost &
- HPI or LPCI non available &
- RPV pressure > 26 bar

result in the action group F 50, „RPV high-pressure injection“. The grading symbol shows that these are Emergency Actions.

Let us have a look at the several meanings of these grading symbols: (Fig. 2/6)

The conspicuously designed grading symbols (Fig. 2/6) fulfil two functions:
1. Eye-catcher to easily find the action group upon the transition from level 2 to level 3 of accident management (consecutive sheets) and

2. Marking whether the actions are
   - manual actions in the design-basis range
   - automatic operating systems
   - automatic Reactor Protection Systems (safety systems) or
   - Emergency Actions.

Through the special marking of the Emergency Actions (completely black square) the operator can also immediately recognise that the safety objective is no longer met or in acute danger of being violated in this situation.

If the operator needs more detailed information for an action group, he has to go to the following pages inside the same register, to level 3:
One action group may consist of several individual actions. These individual actions are listed in level 3 (Fig. 2/7).

For each action the following is given:
- mandatory prerequisites
- the name of the action
- special directions for this action
- for Emergency Actions also the time needed to realize the action.

There may be actions for different tasks, but as well actions for the same task and aim. In the last case the operator has to choose the most adequate action, which is realizable within the required time. For that, the additional information in the annex of each module is helpful. In the last column for each action is the reference where the „step by step programme“ for the action is to be found. These step by step programmes are level 4 of the procedure (Fig. 2/1 and 2/3).

Depending on the type of action, normal action or Emergency Action, the step by step programme is in the Operation Manual (OM) or in the Emergency Manual (EM).

2.3 Additional information in the EOP

In order to facilitate the assessment of the situation and the selection of actions for the operator, the annex of each action module provides additional information in processed form, for instance:

- important, distinct information on the safety parameter, e.g. level/cross-sectional/volume curves with limit marks, important design dimensions etc. Fig. 2/6 shows an example. This information gives orientation and provides technical data for quantitative rough estimates which may be necessary.

- Information on the performance required to maintain the safety parameters in or to return them to the "normal range". Fig. 2/9 shows as an example the required injection mass flow into the RPV to maintain the level, as a function of time following a reactor scram. The information applies to an RPV pressure of 5 bar and feedwater temperature of 300°C.

- Information on the effectiveness of the actions available. This information is particularly important for emergency action. Several emergency action alternatives have often been worked out here and described in the EM, from which the operator has to choose the most appropriate one in each individual case. In Fig. 2/10, the characteristic injection curve of the high-, medium- and low-pressure injection systems into the RPV are shown as examples.
- Variations in time of the safety parameters in question for typical design-basis and beyond-design accidents. These curves permit the operator to estimate the expected time behaviour of the safety parameters and hence to include the factor time in his considerations. These curves also show the maximum or minimum values to be expected of the safety parameters. This information is naturally event-related and therefore applies to very specific prerequisites and a defined sequence only. Nonetheless, it is a valuable information for the operator. One example of this is given in Fig. 2.11. Here, we have deliberately chosen the presentation that the operator finds on the recorders in the control room.

2.4 Handling of the EOP

The EOP has been designed for the management of accidents occurring in the course of power operation.

The event which leads to the application of the EOP is the triggering of a reactor scram or undervoltage or underfrequency on one or several emergency busbars.

Beginning with the aforementioned entry to the Emergency Operating Procedure, the operators have to check at regular intervals during the accident the safety parameters according to Fig. 2/4 and Fig. 2/5 to see whether changes in operating conditions require new actions.

In the design-basis range, particularly within the range of usual incident transients, the operators know the required actions by heart so that it is not necessary to draw on the EOP. These skills can be attained by adequate training (see section 5) and are advantageous for actions that have to taken very quickly.

As a result of the fact that on the first page of each action module (query logic, level 2, see Fig. 2/5) both the entry criteria to the module as well as the criteria for the action to be taken and the actions themselves are specified in a brief text, these pages contain all the information normally required for the procedure. The EOP contains 14 such first pages. If the division of tasks among staff members is taken into account for the allocation of the safety parameters/action modules (we share it at least to 3 licensed operators), and if the fact is taken into consideration that never all action modules are involved during an accident, an easily manageable maximum of three pages arises in the EOP, which have to be handled simultaneously within the sphere of responsibility of one operator.

If the proportion known by heart is deducted from this volume, it can be seen that the EOP is well manageable.

Now that the EOP has been expanded by the Emergency Actions, we made extended tests in order to find the optimum "casting" in the responsibility of operators for individual safety parameters/action modules by role plays on the new KRB II simulator.

3.1 Why Emergency Actions?

Unceasing efforts and the accidents of Harrisburg and Chernobyl sped up the demand to use the above-described reserves of existing operating and safety systems and, in addition, to reduce further the above-mentioned residual risk of core meltdown accidents by selective backfittings.

In case of failure of the highly reliable design-basis operating and safety systems, the Emergency Actions aim to

- prevent damage to or meltdown of the reactor core
- prevent or mitigate impacts on the environment
- reduce unavoidable damage to the plant
- restore the safety and design-basis conditions of the plant and thus
- eventually protect the environment and the personnel from inadmissible radiation exposure.

A preliminary assessment of the efficacy of Emergency Actions showed that they reduce the core meltdown frequency by about one decade. A decisive improvement for PWRs results from the fact that Emergency Actions make it possible to transfer almost all core meltdown accidents from the high pressure path to the low-pressure path. This is of particular importance because a penetration of the containment cannot be excluded with the high-pressure path. Such a penetration would have serious consequences in terms of activity release to the environment.

Including the Emergency Actions, the requirements of the emergency operating procedure are set up such as to result in a "retreat strategy":

**Stage 1**

In the first stage the reactor is to be shut down and the core cooled and protected from destruction and melting.

**Stage 2**

If core melting cannot be prevented, the melt is to be cooled down within the RPV still so as to prevent the meltdown of the reactor pressure vessel (RPV).

**Stage 3**

If the RPV meltdown cannot be prevented, the core melt is to be cooled down in the containment so as to prevent the containment from being destroyed by the melt. During stage 3, all actions are aimed at preserving containment integrity and thus safely hold back activity inventory.
3.2 **What are plant-internal Emergency Actions?**

Emergency Actions are almost exclusively manual actions that serve to set up
- forced operation,
- special operating modes,
- unusual links between operating and safety systems and
- to connect systems especially kept in hand for emergencies.

Various Emergency Actions deliberately put up with certain plant damage, if this is a way to prevent or mitigate damage to the environment.

In contrast to safety measures that have a highly binding character, the instructions for Emergency Actions are a target not a must.

Although the overall objective is to make emergency measures as simple, clear, direct and robust as possible to increase the confidence in them, this is counteracted by the naturally augmented complexity of the processes leading to beyond the design basis (Fig. 1/1).

Some Emergency Actions worked out for the Gundremmingen nuclear power plant (KRB II) require intervention with the Reactor Protection System. This increases the complexity of switching actions.

Design-basis actions are described in the operating manual, Emergency Actions in the Emergency Manual. What actions are to be taken in the individual case, design-basis or Emergency Actions, is laid down in the Emergency Operating Procedure (Operating Manual, part 3, chapter 1).

3.3. **What Emergency Actions have been prepared?**

The following investigations were carried out in order to determine necessary and efficient Emergency Actions: What design-basis safety functions can fail? What is the minimum output absolutely necessary? What is the maximum period after which this output has to be available? And what alternative systems can produce this output? To answer these questions, various accident scenarios were played through with best-estimate calculations.

Thus, for instance, the scenarios of "main steam pipe rupture and failure of all feed and heat removal systems" and "station black-out" were calculated for KRB II. The two model cases are largely covering the entire spectrum of cases to be considered. The objective is for these Emergency Actions to cover a band as wide as possible of failure situations and accident scenarios.
As it is impossible to have a clear and detailed overview of all potential emergency situations, it is expedient to prepare for each protection target several Emergency Actions from which the most suitable can be chosen in the respective case. In this connection, the energy supply for the active components in the emergency action plays a major role. As sequence analyses have been calculated for only a few scenarios, we have taken the practical experience and know-how of our engineers at KRB II (the same procedure was applied in other plants) as a basis to determine suitable measures. At KRB II, however, we gave priority to determining the efficacy of the actions as reliably as possible. Thus, for instance, we have computed the hydraulic resistance's for the entire emergency cooling and residual heat removal system and for the secondary cooling water system and partly secured the results by tests in order to be able to determine the feed mass flows achievable through these systems when applying special operating modes.

Wherever Emergency Actions could be stepped through and tested without damaging the plant, we have done so. Thus, for instance, we created a new lateral connection to link the emergency generator busbars of the two units, supplied one bus via the diesel of the other bus and identified the maximum transmission capacity by load tests.

We have also compiled with the target of practice-oriented Emergency Actions at KRB II in that we had a major part of the Emergency Actions worked out by operators involved in practical operation.

The Emergency Actions which have been eventually developed and prepared for KRB II are presented in section 4. Fig. 3 gives a survey of the contents of the Emergency Manual, part 3.

A probabilistic system analysis (PSA, stage 2) is currently being carried out for KRB II by the Gesellschaft für Reaktorsicherheit (GRS) (Association for Reactor Safety), Cologne, which takes also account of Emergency Actions. It can be expected that the results of this analysis will influence the selection of Emergency Actions to be taken.

3.4 Licensing of Emergency Actions by the authority

The envisaged Emergency Actions are submitted to the authority and their expert for approval. The following factors are investigated:

- compatibility with the safety concept of the design,
- efficacy and
- feasibility.

A formal modification application has to be filed for any alterations to the licensed plant.
3.5 Initiation of Emergency Actions

The criteria for initiating Emergency Actions are laid down in the "Emergency Operating Procedure", Operating Manual, part 3, chapter 1 (see section 2).

The Emergency Actions are allocated to the categories A or B.

Actions falling under category A are actions that
- have to be taken quickly in order to be effective (< 2 h) and
- have to be taken prior to imminent accidents beyond the design
- basis in order to prevent their occurrence, to halt their
development or to mitigate their consequences.

The decision for category-A actions lies in the hands of the shift supervisor.

Actions falling under category B are actions
- for the triggering of which sufficient time is available
- that have direct impacts on the environment
- that have considerable impacts on the plant or
- that are subject to extraordinary risks.

The decision for category B actions basically lies in the hands of the Emergency Response Organization. The regulatory authority has reserved the right to decide in individual cases.

3.6 Implementation of Emergency Actions

Emergency Actions are carried out by:

- deputy shift supervisor
- reactor operator
- control room operator

- shift fitter
- shift electrician
- firemen
- craftsmen in the transport department

in the control room

in the plant

The shift supervisor determines the personnel to be employed and the supervisor on site.

If the shift supervisor demands operating teams outside the shift, these are called up by the Emergency Response Organisation (ERO). Upon the occurrence of accidents, the ERO is convened directly by the manager on call or by the shift supervisor (Fig. 2/1).
3.7 Implementing instructions for Emergency Actions

The step-by-step programmes for implementing Emergency Actions are written down in the Emergency Manual. They are similarly structured to the step-by-step programmes for implementing actions in the design basis as laid down in the operating manual, part 4 (system operation).

3.8 Set-up of Emergency Manual

The set-up of the Emergency Manual is illustrated in Fig. 3/1. Part 2 describes the shutdown via partial control points (emergency control rooms), part 3 describes the Emergency Actions and part 4 describes the direct control of components in the switching stations.

3.9 Scope of Emergency Manual

The Emergency Manual consists of three files and comprises the instructions for 94 Emergency Actions per unit. 54 instructions alone were necessary to describe the actions to be taken for power supply restoration in the emergency power and auxiliary power busbars. The large number is a result of the variety of possible interconnections within one unit and between the two units.

3.10 Structure and contents of implementing instructions (step-by-step programmes)

The structure of the implementing instructions was standardised for the whole of Germany by a working group of operators.

The implementing instructions contain the following information (illustrated in German in Figures 3/2 - 3/11):

- The target pursued by the action
- The indispensable prerequisites to carry out the action
- The personnel and time requirements to carry out the action. These data are classified according to groups of persons that perform the same subtask at the same time. These subtasks are referred to as "job blocks". These job blocks are marked on the right hand side of the following step-by-step programmes. Each job block has its time requirements stated.
- Performance data, efficiency of the action (pertinent graphs attached in the annex)
- Concise and clear text description of the action
Logic for parallel execution of action steps. This logic is of great advantage as it offers the shift supervisor or action co-ordinator a swift overview of the required interlocking of the job blocks and contains all essential data on the job blocks. The logic is not used if the jobs to be carried out are serial ones.

- Step-by-step programmes for individual jobs. Here, the presentation is the same as in the operating manual, part 4.

- Criteria for terminating the action

- If necessary, step-by-step programmes for resetting

- Annexes containing
  - performance curves for the action
  - function diagrams with marked switching path
  - plans of sites and buildings showing line connections etc.
  - photos of essential components and identification of intervention points.

- Lending copies
Copies of the relevant step-by-step programmes are kept ready for quick withdrawal for each job block. The lending copies have a cover page which can be used to note down additional information regarding the job block in question, such as

  - necessary auxiliary equipment and its location
  - special indications as to the execution of individual action steps.

By keeping these copies ready for withdrawal, a time gain is achieved.

3.11 Instrumentation

The development of the Emergency Actions has shown that in some cases the existing instrumentation is not sufficient to cover adequately the areas beyond the design basis. Here, retrofitting was performed.
4. Prepared Emergency Actions and already installed and planned backfitting Projects

4.1 Prepared Emergency Actions

In Fig. 4/1 to 4/6 all Emergency Actions which can be done now at Gundremmingen are listed. The achievable performance, the needed personal and time requirements are indicated.

The points of main effort are reactor and containment feeding resp. flooding and to ensure auxiliary and emergency power supply.

4.2 On Site Switch of Components

For unexpected events there must be assumed failures or outages of instrumentation and control devices.

In order to be able to switch in these cases the required components on site we have carried out special Emergency Actions.

These actions are prepared for 10 kV- and 660-V-motors and the emergency diesels. Current the tools to switch the various types of 380-V-motors are being made.


4.3 Backfitting

The KRB II plant features were designed in the early seventies. As one of the modern light water reactors KRB II corresponds to actual state of the art in the field of nuclear safety standards in the eighties.

Backfitting in nuclear power plants are discussed all over Germany except the so called "Konvoi"-series. They can have different reasons and can be classified into four divisions:

A) Backfitting or plant updating to current licensing requirements

The plant is in a state of fulfilling all the current licensing requirements without any relevant modification.

B) Taking on account of plant safety margins (Recommendation of the Safety Advisory Commission, RSK)

These actions should prevent severe nuclear damages.
- Connexion between condensate pumps and feedwater system (F61) (Fig. 4/6).
This connexion enables RPV injection during loss of feedwater system and high pressure injection system and supports the pressure in the feedwater system (prevents evaporating).

(ready 1991)

- Diesels busbars coupling (S80).
The coupling provides the possibility to use any of the 10 diesels for feeding any busbar in each unit (A, B, C).

(ready 1988)

- Connexion between nuclear service cooling water system and RPV injection/residual heat removal system (F74, F75, X41, X111, X81, X112) (Fig. 4/7).
This connexion feeds river water into the reactor or into the containment.

(ready 1988)

Connexion between fire protection system (F77), nuclear service cooling water system VE20 (F75, X112), condensate storage tank (W24), poison solution tank (W34).

(ready 1990)

C) Internal Emergency Actions
(Recommendation of the Safety Advisory Commission, RSK)

These actions should mitigate severe nuclear damages

- Containment venting (V11) (Fig. 4/8 left)
The filtered venting prevents overpressure in the containment.
The objective is to avoid venting completely.
Some investigations were done to estimate the ultimate pressure limit, the KRB II containment can withstand. A figure of 10 bars was found out for the concrete structure and the main steel parts. Some finite element (FE) calculations and some minor component changes extended the ultimate pressure limit to about 10 (145 psi) bar for the containment in total.

(ready 1990)

- Inerting Wet well Air (Fig. 4/8 right).
During core melt down the Zirconium of the fuel rod clad are able react with steam. Oxidation of the cladding takes place, thus producing Hydrogen.
The air of the wet well is inerted by Nitrogen (passive measure) to prevent explosion.

(ready 1990)
- Coping with the Hydrogen Problem (active measure).
The dry well is not inerted, therefore a rest of Oxygen remains in the containment.
Hydrogen must be removed either by catalysator by ignition before the concentra-
tion for explosion is reached. The reactor safety comission- evalulations about this
point are not settled.

- Control Room Venting (V12)
There is a back up air supply system to minimize the radiation exposure of the
shift personal. It keeps the control room in overpressure by filtered fresh air input.
It is intended to connect this system if the containment venting system is in opera-
tion.

(ready 1990)

- Underground cable connexion to the power grid to cope with the incident "Station
black Out" (S90).

(ready 1991)

- Additional possibilities for taking samples and specific equipments to measure
nuclides. Both devices are still in discussion.

D) **Backfitting due to PSA consequences**

- Functional Separation of One Reactor Injection/Heat Removal System, RHR
(F56).
There are two pumps, one each for high pressure and low pressure. Both have a
common cooling water system. That's a reason for common mode failure. Thus
adding a separate cooling to the high pressure pump, it will become completely in-
dependent from the low pressure pump.

(ready 1991)

- Increased Redundancy and Diversity for Reactor Injection/Heat Removal System
"ZUNA".
Due to common cause failure consideration a expensive new system was added
to the plant as a 4th redundancy and as a diversity to other RHR systems. These
considerations led to a system similar to the 3 existing systems (Fig. 4/9), but with
completely different components (pumps, valves, heat exchangers, diesels, con-
trol circuits and batteries).
This system is independent from river water.

(ready 1995)
5. **Training in the use of the EOP and in processing of the Emergency Actions and Experiences gained with the application**

   (Authors: A. Leinauer, Manager of the Training Department and W. Stadelmann, Deputy Manager of the Training Department)

5.1 **Introduction**

The Regulations from 1990 of the Federal Ministry of Environment determine that the responsible shift personnel have to absolve a periodical refresher training.

These regulations also determine that emergency management procedures must be included in this refresher training. The emergency management procedures which are retrained include:

- the use of the Emergency Manual
- event oriented procedures
- symptom based procedures (EOP)
- practical training in the emergency control room and on the site
- practical and theoretical training on the simulator

The practical and theoretical training is laid down in a three yearly refresher programme. This programme has to be submitted to the licensing authority for evaluation. The licensing authority also controls the retraining records on a yearly basis.

In addition to the above mentioned governmental regulations, the NPP Gundremmingen KRB II has given itself internal regulations in regard to the retraining of Emergency Operating Procedure, Emergency Manual, Emergency Actions and the co-operation between the various parts of the management for severe accidents. This retraining occurs in a three yearly cycle.

We intend to enhance the frequency of this training and practice to a yearly cycle.

The participants of this programme are:

- Shift supervisor
- Reactor Operators
- Unit Managers
- Technical Management
- Emergency Response Organisation
- Management of the support departments and chemical surveillance and maintenance

The job blocks described within the Emergency Manual are retrained once a year. In this case the shift electricians and mechanics as well as the NPP internal fire fighting brigade are involved.
- Diversified RPV Safety/Relief Valves.
  Based upon common cause failure considerations and due to the great importance
  of keeping RPV pressure within safe limits, RSK stated the recommendation of di-
  versified safety/relief valves. To incorporate this in the plant, KRB II installed
  3 small valves of 25 kg/s flow each (compared with 11 main valves of 150 kg/s
  flow each).
  (ready 1993)

- Diversified RPV Level Instrumentation.
  The present level instrumentation has a high redundancy but no diversity. As a
  safety parameter of high importance we installed a diversified measuring installa-
  tion. We are testing some sort of level switches operating on the basis of electro-
  magnetic coupled ultrasonic sensors (EMUS, developed by KWU). If the qualifica-
  tion by the independent experts is achievable is uncertain.

  A third diverse countermeasure is an in core temperature instrumentation, which
  we had installed already. We gained good experiences with this device.
  (ready 1992)

- Diversified Isolation Valves
  Considering the direct cycle of modern BWRs the reliability figures for isolating the
  primary containment lines are of high importance. This belongs to the main steam
  isolation valves as well as to the feedwater isolation valves. What we are consider-
  ing is a reduced surveillance testing interval or a third and diversified isolation
  valve for each of the main containment lines.

4.4 PSA Improvement as consequence of backfitting

The world-wide discussed containment venting and inerting raises the safety by an in-
considerable amount.

But the following backfitting decreases the PSA results by factor 2:

- Separation of high pressure pump from low pressure pump in one RHR system.

- Connexion between condensate pumps and feedwater system.

- Diversified safety/relief valves.

The new RHR system (ZUNA) achieves a significant reduction of the PSA result by
factor 11.
- Getting into the emergency control room under specified accident circumstances e.g. and Heat Removal System Compartment Level high simulated.

Emergency Diesel generator operation during LP Heat Removal System operations. The Diesel generator start will be actually tripped, causing a restart of the Heat Removal System.

- Training in the recognition of the signalisation in the electronic racks of the emergency control room when the reactor safety system is in action.

- Start up of the emergency diesel generators by hand when the automatic restart fails (Fig. 5/6).

- Operation of systems from the emergency control room e.g.
  + LP Heat removal system
  + Spent fuel storage cooling system.

- Override of the Reactor Protection System (Fig. 5/7, 5/8 and 5/10)

- Refilling of storage tanks (Fig. 5/9)

- Manually direct switch of components in the switch gear building (overriding of the control device) (Fig. 5/10).

5.4 Training of the Emergency Operating Procedure (EOP) on the simulator

The operators who have to apply the EOP must undergo detailed training in the structure, systematic and approach of the EOP. This training can be provided in classrooms as mentioned before.

Indispensable, however, and by far the most important part of this schooling is in-depth training on a simulator which is a true replica of the real plant without any differences. This simulator training is essential to practise the role play and to develop the required familiarity and skills of the operators.

Fortunately we have got a very sophisticated full scope simulator in the mid of 1993 which represents our plant in a very detailed manner. It is located in the simulator center of the Nuclear Power Simulator Community (KSG) in Essen.
The NPP Training Department is responsible for all the emergency management training.

5.2 Theoretical training in the NPP Training Center including Computer Based Training

The theoretical training is being done in the NPP Training Center during the one week day shift. Each day shift has a preplanned training programme. Within this training programme certain chapters of the Emergency Manual are discussed. The discussions are lead either by qualified trainers from the Training Department or by the several shift supervisors.

The theoretical training of the emergency management procedures also includes the dry training on site e.g. in the switchgear building, emergency diesel generator etc., without any actual switching or operational actions. During the theoretical training it is also very important to familiarize the participants with the on-site hardware e.g. switchgear, access possibilities to the emergency control and diesel generating rooms. This knowledge has to be updated and refreshed. In this connection the emergency management procedures concerning actions in the emergency control room are discussed in the emergency control room itself in order to have the direct reference to the hardware.

The governmental refresher training regulations from 1990 also allows the use of Computer Based Training (CBT - with a normal Personal Computer for self-studies). Until now we have implemented the containment venting system as CBT. For 1991 we are planning to implement the reactor power control system, the Emergency Operating Procedures and two chapters from the Emergency Manual.

We expect that due to the interaction of CBT, the learning and retention of knowledge during self-studies will be more efficient and will lead to smoother operation during normal and emergency plant operation. The lesson learned so far shows, that CBT is helpful in certain fields but not in general. It is useful for a first introduction in a process or for some retraining tasks. It is no substitute for training with a lecturer.

5.3 Practical training on the NPP site

During the day shift refresher training programme, a certain amount of practical Training in connection with the Emergency Manual is also included (Fig. 5/1 - 5/10).

This practical training actually involves the switching of plant hardware for emergency purposes without disturbing the normal plant operation.

The practical training covers such cases as:

- Feeding Danube water via the low pressure heat removal system into the Reactor Pressure Vessel with the help of the mobile fire fighting pump. In this case all the necessary actions up to the heat removal system are actually started. Here the internal fire fighting brigade is also being trained (Fig. 5/1 - 5/5).
As soon as possible we started to take tests to prove the practical topics:

- the general practicality of the approach
- the best appropriate casting for the members of the shift crew
- the best way to apply, to conduct and to process the EOP (priorities, handling, cooperation)
- to find out possible improvements and further developing potential of the EOP
- to prove the information display in the main control room with respect to the application of the EOP

To follow these aims we carried out 22 tests at the simulator. Most of the simulated accidents escalated to beyond basis conditions.

Essential findings were:

- The procedure approach works in an outstanding manner.
- Due to the consequent application of the EOP no safety relevant measures were failed to do.
- The chosen layout of the EOP is practical.
- The shift crew considers the EOP as a real help to cope with accidents.
- Because of the application of the EOP the accidents are managed more steady, more systematically and better organized.
- Since we have not got the safety-objective information system yet we have to improve the instrumentation display of the main control room at many points.

To give you a few examples:

We marked all instruments showing safety parameters needed for the processing of the EOP by a red frame.
We stored all 2 out of 3 triggers of the Reactor Protection System and indicated it by lamps on the Reactor Protection System Panel. (There are a lot of very short lasting triggers being important to check the plant status.)

As a result of the tests we consider the following casting as best appropriate:

a) The shift supervisor is not directly involved in the processing of the EOP. He remains in the background, watching the sequence of the event and asks for missing information. He makes the decision for important measures.

b) The deputy shift supervisor checks the Reactor Protection Safety Panel, the power busses and the activity instruments. He co-ordinates the operation of the shift craftsmen on site.

c) The operator at the Reactor control desk watches the essential parameters and functions: undercriticality of the reactor, reactor level, reactor pressure, reactor feedwater supply and heat sink.
d) **The operator at the auxiliary system control desk** watches the nuclear and conventional auxiliary systems, the substitute heat sink (suppression chamber) and the containment isolation valves.

Announcements, mutual information:

As a further result of the tests we decided the shift members should announce in general only deviations from the expected sequence. Some exceptions from this rule were made for information’s being important for all participants.

During the first half of 1994 all shift crews of our plant got a one week crash training in the application of the EOP at the new simulator.

Due to the fact, that the EOP comprises also grave measures which decision making is up to the Emergency Response Organisation (Crisis Management), the members of this organisation have to undergo a training sessions on the EOP also. They confirm their ability to handle the EOP by emergency practice.

The following summary shows the range of beyond design accidents which are realized on the simulator and the Emergency Actions which are necessary in order to ensure that the NPP stays within the safety objectives:
<table>
<thead>
<tr>
<th>Safety Objectives</th>
<th></th>
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</thead>
<tbody>
<tr>
<td>Beyond design accidents which are realized on the simulator</td>
<td>Emergency Actions to ensure the safety objectives, which can be trained on the Simulator</td>
</tr>
</tbody>
</table>

<table>
<thead>
<tr>
<th>Subcriticality</th>
<th></th>
</tr>
</thead>
<tbody>
<tr>
<td>- Failure of the scram system</td>
<td>- Operation of the poisoning system</td>
</tr>
<tr>
<td>- Failure of any control rod drive (electrically or hydraulically)</td>
<td></td>
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</tbody>
</table>

<table>
<thead>
<tr>
<th>Core cooling / Heat sink</th>
<th></th>
</tr>
</thead>
<tbody>
<tr>
<td>- Failure of the RPV pressure release system</td>
<td>- Bridging of reactor safety System Signals in order to reactivate the normal operational heat sinks (e.g. withdrawal of steam via the main or auxiliary paths). Operation of the reactor water purification system.</td>
</tr>
<tr>
<td>- Complete loss of water from the suppression pool</td>
<td>- Flooding of the containment with river water</td>
</tr>
<tr>
<td>- Failure of the pressure suppression system due to an open backstroke valve by loss of coolant</td>
<td>- Pressure reduction through the Containment-Spray-System and the Suppression-Pool-Spray-System</td>
</tr>
<tr>
<td>- Failure of all heat removal and feed systems (3 x 100 %)</td>
<td>- Reactivation of operational feedwater systems via bridging of safety signals</td>
</tr>
<tr>
<td></td>
<td>- Replenishment of the water inventory in the systems from the fire fighting water network (river water) through the activation of simulated manual operating valves (Feeding at different points).</td>
</tr>
<tr>
<td></td>
<td>- Simulation of mobile feed pumps in different systems.</td>
</tr>
<tr>
<td></td>
<td>- Feeding the RPV with the condensate pumps (Bypassing the feedwater tank)</td>
</tr>
<tr>
<td>Integrity of the containment and activity Barriers</td>
<td></td>
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<tr>
<td>------------------------------------------------------------------</td>
<td>------------------------------------------------------------------</td>
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<tr>
<td>- Failure of the isolation valves of the containment</td>
<td>- Closing of the isolation valves through simulated manual operation in the plant locality</td>
</tr>
<tr>
<td></td>
<td>- Pressure limitation of the containment through the venting system</td>
</tr>
</tbody>
</table>

<table>
<thead>
<tr>
<th>Electrical Power Supply</th>
<th></th>
</tr>
</thead>
<tbody>
<tr>
<td>- Failure of the plants auxiliary and emergency electrical power supply through a combination of any single failure</td>
<td>- Interconnection of separated Redundancies through cancelling of Interconnection locks (e.g. by emergency power lines)</td>
</tr>
<tr>
<td>- Time related exhaustion of the batteries with the corresponding failures</td>
<td>- Feeding of the RPV without pumps, only with help of the available pressure energies (e.g. feedwater tank)</td>
</tr>
<tr>
<td></td>
<td>- Manual start-up of the emergency power diesel generators</td>
</tr>
<tr>
<td></td>
<td>- Operation of power switches directly in the switchgear building (in case of malfunction of instrumentation or control devices).</td>
</tr>
</tbody>
</table>
6. Instrumentation Problems

It is an indispensable prerequisite to apply the EOP successfully to have a concentrated and clearly arranged instrumentation. We do not have the ideal configuration yet.

A few instrumentation shortcomings are mentioned in section 5.4 where the experiences are reported gained during the test of the EOP at the simulator.

6.1 Safety Parameter Display System (SPDS)

Part of the original SIEMENS design is a panel showing all trigger signals, the logic and the trip signals of the Reactor Protecting System (RPS) and the feedback signals from the launched components of the safety systems.

It is a very helpful instrumentation to check the function of the RPS and the tripped safety systems.

As a result of the tests of the EOP we stored all 2 out of 3 triggers of the RPS and indicated them by lamps because a lot of them are very short lasting ones. In former time they were just indicated as long as they were actuated.

Due to the fact, that the RPS panel shows not all safety parameters and other parameters needed to handle the EOP and in order to get an overview about the plant we designed parallel to the construction of the two new units B and C an overview panel additional to the original SIEMENS instrumentation of the control room (Fig. 6/1). It shows the main parameters and the state of the steam/water cycle, the reactor and the safety systems including the conditions inside of the containment, reactor building and turbine hall. Unfortunately it was not kept up to date when the emergency measures were implemented in 1990. Because it was not part of the licensing procedure, we have not got the approval to use it in full scope.

Currently we are making the first steps to design a new process computer. It is scheduled to replace the old one in 2001. We intend to take the chance and implement a special Safety Parameter Display System with the new computer. We assume that 4 screens will be sufficient to show all the needed information's on a SPDS. Fig. 6/2 shows a draft of the SPDS screen of the safety objective „Reactor Core Cooling“ (unfortunately in German).

6.2 Other instrumentation problems

- International well known are the problems with the reactor level measurement instrumentation working with differential pressure. We make efforts to get diverse level measurement instruments (see section 4.3).
- We intend to install a gas sample system for the containment to measure high level activities during severe accidents and to determine the various nuclides. The current equipment is not sufficient for this extreme demands. We are in negotiations with SIEMENS to buy their „In-Situ-Poolprobenehmer“.

- The neutron flux in the core of the KRB II reactors is measured by 5 neutron flux monitoring rods. Each of them is able to measure a distance of 6 control rod cells only. This measurement equipment is sufficient in the main control room to check the subcriticality where all 5 monitoring rods are available.

We have some uncertainties to check the subcriticality in our two notstand control rooms due to the fact, that in each of them solely one neutron flux rod is monitored.

7. **Critical decisions, uncertainties, remaining open questions**

7.1 **Containment flooding**

Similar to the Swedish BWR we have prepared an Emergency Action which enables us to flood the reactor cavity. We intended to launch this action in case we expect the RPV will melt down. The water in the cavity should cool the melt and the debris. Due to the again arising discussion in Germany of water steam detonation we cancelled this action in our EOP. We are rather sure it will not have serious effects to the containment and the safety equipment but we do not dare to apply the action until this assumption is proved.

7.2 **Recriticality during reflooding of the uncovered core**

The CORA tests in the Nuclear Research Centre (KfK) Karlsruhe, Germany, (Part of the „Programme Nuclear Safety, PNS“) showed that in case of an uncovered core the control rod blades will melt down a considerable period (> 1 h) before the fuel bundles crash down or melt down. If the core will be flooded within this period recriticality may occur. SIEMENS evaluated, that depending of the feedwater mass flow, tremendous power (to 5 times full power) and partially destruction of the core may be the result.

We are uncertain that we are able to control this reflooding sufficiently by control of RPV level and feedwater mass flow or by an early injection of boron to the core. The capacity of the boron poisoning system is big enough but there are uncertainties about the distribution of the boron.
7.3 Operation of the containment spray system in case of severe accidents

The high level of hydrogen produced during severe accidents results to problems with the operation of the containment spray system. The hydrogen concentration must be carefully watched to avoid the exceeding of the ignition limits while operating the spray system.

8. Answer to the questionnaire of the Programme Committee of the Specialist Meeting (as far as not answered in the items above)

(f) We have no computerized operator aids for direct operation support applied so far.

(g) In case of a severe accident, we are changing the organization and installing the Emergency Response Organization (see section 1).

(k) All preparations for severe accidents have been approved by the authority and her independent expert (TÜV). This includes

- EOP
- Emergency Actions
- Emergency Manual
- Emergency Organization

The authority is involved in emergency practices also.

We are convinced, due to all the efforts mentioned in this paper, to have made a big step forward to minimize the residual risk.
### Strategy to prevent, to control and to mitigate accidents within and beyond design basis

<table>
<thead>
<tr>
<th>normal operating</th>
<th>1. Thorough and conservative design with large safety margins</th>
</tr>
</thead>
<tbody>
<tr>
<td></td>
<td>2. Thorough tests and inspections to guarantee and maintain quality</td>
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<td></td>
<td>3. Carefully developed and elaborated manuals and documentation</td>
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<tr>
<td></td>
<td>4. Highly qualified staff</td>
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<tr>
<td></td>
<td>5. Intensive training of the staff</td>
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<td></td>
<td>6. Clearly arranged instrumentation</td>
</tr>
<tr>
<td>deviation from normal operating</td>
<td>7. Selfstabilizing behaviour of the systems</td>
</tr>
<tr>
<td>accidents within the design basis</td>
<td>8. Operational control systems and limitation systems</td>
</tr>
<tr>
<td>accidents beyond the design basis</td>
<td>9. Reactor protection system and safety systems</td>
</tr>
<tr>
<td>beyond the design basis</td>
<td>10. Flexible using of the operational systems, the safety systems and the components with their real capabilities, if necessary beyond their originally designed function</td>
</tr>
<tr>
<td></td>
<td>11. Upgrading or backfitting of systems for emergency actions</td>
</tr>
<tr>
<td></td>
<td>12. Symtombased approach to control accidents. Emergency Operating Procedures (EOP’s)</td>
</tr>
<tr>
<td></td>
<td>14. Emergency Response Organization to support the operating staff and to correspond with the off-site organizations</td>
</tr>
<tr>
<td></td>
<td>15. Training of the Emergency Operating Procedures and of the processing of the Emergency Actions</td>
</tr>
</tbody>
</table>
Fig. 17-2: Four safety levels - defence in depth

- Design range: cladding temperature < 1200 °C
- Core damage within RPV
- Core molten in containment

Probability distribution:
- Core melt probability:
  - $10^{-7}$
  - $10^{-6}$
  - $10^{-5}$
  - $10^{-4}$
  - $10^{-3}$
  - $10^{-2}$
  - $10^{-1}$
  - $10^{0}$

- Normal operation
- Events
- Design basis accidents
- Accidents beyond design

- Complexity of Phenomena

Design basis Operating Manual
Beyond design basis emergency manual
Emergency Response Headquarters
- Official contacts with authority, local government, police
- Contact with SIEMENS crisis team

Staff Operation
- Plant Operation
- Make up water, Watertreatment, Decontamination

Staff Radiation Control
- Radio-Chemistry
- Radiation Monitoring
- Health Physics
- Environmental Lab
- Environmental Sampling
- Environmental Monitoring

Staff Special Tasks
- Fire Brigade
- Ambulance
- Maintenance, storekeeping

- Communication Links
- Site Security
- Documentation Provision
- Public Information
Fig. 2/1 - Overview of Event Tasks and Procedures

- Plant Management
  - Plant supervisor
  - Manager on call

- Shift Personal

- Event
- Deviation from normal operation

- Reactor trip

- Post trip check
  - NOVA

- Emergency Operating Procedures (EOPs)
- Symptoms oriented

- Induction of ERO
  - Analysis of event and assessment of plant status
  - Decision of ultimate emergency actions
  - Support of shift personnel

- Normal operating actions

- Operating Procedures (OM)
  - Normal operation
  - Event oriented

- Operating Procedures (OM)
  - Accidents oriented

- Operating Procedures (OM)
  - Beyond design

- In design

- Parallel

- Step by step programme

OM = Operation Manual
* Marked criterias has to be announced to be shift supervisor and to the other members of the shift crew.
(All another criterias should be announced in case of a deviation from the normal expected behaviour only)

<table>
<thead>
<tr>
<th>parameter</th>
<th>query criterias / hints</th>
<th>to announce</th>
</tr>
</thead>
<tbody>
<tr>
<td>neutron flux (reactor shut down)</td>
<td>- scram triggered?</td>
<td>*</td>
</tr>
<tr>
<td></td>
<td>- all control rods inserted?</td>
<td></td>
</tr>
<tr>
<td></td>
<td>- flux monitor at zero?</td>
<td></td>
</tr>
<tr>
<td></td>
<td>- recirculation pumps at low speed?</td>
<td></td>
</tr>
<tr>
<td>RPV pressure</td>
<td>- current value / tendency?</td>
<td>*</td>
</tr>
<tr>
<td></td>
<td>- main steam valve isolation?</td>
<td>*</td>
</tr>
<tr>
<td></td>
<td>- auxiliary steam valve isolation?</td>
<td>*</td>
</tr>
<tr>
<td></td>
<td>- turbine bypaß blocked?</td>
<td>*</td>
</tr>
<tr>
<td></td>
<td>- safety/relief valve working?</td>
<td>*</td>
</tr>
<tr>
<td></td>
<td>- feedwater storage tank sustain steam in operation?</td>
<td></td>
</tr>
<tr>
<td>RPV level</td>
<td>- current value / tendency?</td>
<td>*</td>
</tr>
<tr>
<td></td>
<td>- main feedwater system working?</td>
<td>*</td>
</tr>
<tr>
<td></td>
<td>- high pressure injection system actuated?</td>
<td>*</td>
</tr>
<tr>
<td>Later queries</td>
<td>- mechanical insertion of the control rods completely finished?</td>
<td>*</td>
</tr>
<tr>
<td></td>
<td>- neutron flux monitoring system source range and intermediate range works properly?</td>
<td></td>
</tr>
<tr>
<td></td>
<td>- turbine trip actuated?</td>
<td></td>
</tr>
<tr>
<td></td>
<td>- generator circuit breaker off?</td>
<td></td>
</tr>
<tr>
<td></td>
<td>- feedwater pump gate valves (pressure side) closed?</td>
<td></td>
</tr>
</tbody>
</table>
### Safety Objective and Parameter Table

<table>
<thead>
<tr>
<th>Safety Objective</th>
<th>Safety Parameter</th>
<th>Limit</th>
<th>Action Module</th>
<th>Code</th>
</tr>
</thead>
<tbody>
<tr>
<td>Subcriticality</td>
<td>Control rod position</td>
<td>not all control rods inserted</td>
<td>subcriticality</td>
<td>U</td>
</tr>
<tr>
<td>Core cooling</td>
<td>RPV - level</td>
<td>&lt; 13.91 m (LT 1) &gt; 15.28 m (LH 1)</td>
<td>RPV - level control</td>
<td>F</td>
</tr>
<tr>
<td></td>
<td>RPV - pressure</td>
<td>&gt; 74 bar</td>
<td>RPV - p limiting</td>
<td>D</td>
</tr>
<tr>
<td></td>
<td>Fast RPV pressure drop</td>
<td>RPS - trigger</td>
<td>RPV - suppression</td>
<td></td>
</tr>
<tr>
<td>Heat sink</td>
<td>Suppression pool - temp.</td>
<td>&gt; 32 °C</td>
<td>Supp.pool cooling</td>
<td>K</td>
</tr>
<tr>
<td></td>
<td>Suppression pool - level</td>
<td>&lt; 1.95 m, &gt; 2.05 m</td>
<td>Supp.pool level cont.</td>
<td>N</td>
</tr>
<tr>
<td>Integrity Containment and Activity Barriers</td>
<td>Primary, containment-pressure</td>
<td>&gt; 15 mbar</td>
<td>Coping with LOCA</td>
<td>L</td>
</tr>
<tr>
<td></td>
<td>Secondary containment-pressure</td>
<td>&gt; 5 mbar (P 20)</td>
<td>Management of LOCA</td>
<td>L</td>
</tr>
<tr>
<td></td>
<td>Turbine building - pressure</td>
<td>&gt; 5 mbar (P 30)</td>
<td>Ensure isolation</td>
<td>C</td>
</tr>
<tr>
<td></td>
<td>Isolation valve trigger</td>
<td>RPS-trigger</td>
<td></td>
<td></td>
</tr>
<tr>
<td></td>
<td>H₂ in containment (DW, WW)</td>
<td>&gt; 3.5Vol% &gt; 0.5Vol%</td>
<td>Coping with H₂</td>
<td>H</td>
</tr>
<tr>
<td>Limitation</td>
<td>Activity stack</td>
<td>common Alarm</td>
<td>Limitation</td>
<td>A</td>
</tr>
<tr>
<td>Activity</td>
<td>Activity 00TL Ventilation</td>
<td>common Alarm</td>
<td>Activity</td>
<td></td>
</tr>
<tr>
<td>Release</td>
<td>Activity 00TL rooms</td>
<td>common Alarm</td>
<td>Release</td>
<td></td>
</tr>
<tr>
<td>Water supply</td>
<td>Activity systems</td>
<td>common Alarm</td>
<td></td>
<td></td>
</tr>
<tr>
<td></td>
<td>Feederwater tank level</td>
<td>&lt; 1.5 m</td>
<td>Water supply</td>
<td>W</td>
</tr>
<tr>
<td></td>
<td>Condensate tank level</td>
<td>&lt; 1.0 m</td>
<td></td>
<td></td>
</tr>
<tr>
<td></td>
<td>Poisson tank level</td>
<td>&lt; 2.0 m</td>
<td></td>
<td></td>
</tr>
<tr>
<td>Fuel pool cooling</td>
<td>Fuel pool level</td>
<td>&lt; 11.50 m</td>
<td>Fuel pool cooling</td>
<td>B</td>
</tr>
</tbody>
</table>

**Power Failure**

- 1 Diesel busbar

**Auxiliary and Emergency Power Supply**

- Undervoltage or underfrequency Diesel busbar
  - Uₚ < 0.8 Uₚ
  - Frequency < 47.2 Hz

**Power Supply**

- Power supply S
Fig. 2/5 - Level 1 EOP, action module
RPV level control

Safety Objective: Core Cooling
Action Module: RPV level control

RPV = Reactor Protection System

- \( \geq 1 \) HPI or LPCI feed RPV
- Trend RPV-L increasing
- \( \geq \) steamline ISO-V open

RPS trip LOCA:
- P10 or LT3 trip

- RPV-level 
  - \( > 15.60 \text{ m} \) (LH3)
  - P10 oder LT3 not triggered

- RPV level 
  - \( > 15.45 \text{ m} \)

- RPV level 
  - \( > 15.28 \text{ m} \) (LH1)

- RPV level RDB 
  - \( > 14.50 \text{ m} \) (LH2)

Start Criteria:
- RPV level 
  - \( > 15.28 \text{ m} \) (LH1)
  - \( < 13.91 \text{ m} \) (LT1)

- Increasing
  - Desired level = 14.70 m

- Decreasing
  - Feedwater system operating

- RPV level 
  - \( < 13.00 \text{ m} \)

- RPV level 
  - \( < 12.35 \text{ m} \) (LT2)

- RPV level 
  - \( < 11.00 \text{ m} \) (LT3)

- RPV level 
  - \( < 9.00 \text{ m} \)

- Duration
  - \( > 200 \text{s} \)

- RPV pressure 
  - \( > 26 \text{ bar} \)

- RPV feeding by operational
  - Automatic HPI-start

- RPV flooding
  - LOCA actions

- RPV depressurization by SRV

- RPV feeding HP by emergency actions

- RPV feeding LP by emergency actions

- Start poison-syst.
  - Drywell (Cat.B-action)
<table>
<thead>
<tr>
<th>Grading Symbol</th>
<th>Fulfilling of Safety Objectives</th>
<th>Demanded Actions</th>
</tr>
</thead>
<tbody>
<tr>
<td></td>
<td>Safety objective achieved</td>
<td>No operator actions demanded</td>
</tr>
<tr>
<td></td>
<td>Safety objective achieved.</td>
<td>Supporting operator actions for prevention and mitigation recommended</td>
</tr>
<tr>
<td></td>
<td>Beyond normal range</td>
<td>Normal automatic operation. Recommended supporting operator actions</td>
</tr>
<tr>
<td></td>
<td>Safety objective achieved.</td>
<td>Reactor protection systems should trip. Recommended supporting operator actions</td>
</tr>
<tr>
<td></td>
<td>Beyond normal operation limits</td>
<td></td>
</tr>
<tr>
<td></td>
<td>Safety objective achieved.</td>
<td></td>
</tr>
<tr>
<td></td>
<td>Beyond reactor protection limits</td>
<td></td>
</tr>
<tr>
<td></td>
<td>Normally safety objective</td>
<td>Accident management actions required</td>
</tr>
</tbody>
</table>
RPV feeding by high-pressure systems:

Choose the actions depending of availability and the required mass flow. Parallel try to restore the regular safety systems.

(sequence according to the priority)

+ Feedwater Level > 0,5m
+ REACTIVATE THE FEEDWATER PUMPS BY OVERRIDING OF THE COMPONENT PROTECTING SYSTEM

+ Condensate tank LEVEL > 1,0m
+ REINFORCED OPERATION OF CONTROL ROD DRIVE FLUSHING WATER SYSTEM YT

+ Condensate tank level > 1,0m
+ REINFORCED OPERATION OF RECIRCULATION PUMP SEALING WATER SYSTEM TE

General direction:

If you are not able to stop the sinking of RPV level, try to lower RPV pressure by using the Safety Relief Valves (→ D120) and feed additionally with Low pressure injection systems (→ F60 / F70)

+ OFF-SITE EMERGENCY ALERT NECESSARY?

≥ Subcriticality reactor, in long term

CHECK

safe final condition

to which the plant has to be brought (depending on this actions and situation)
Erforderlicher Einspeisemassenstrom RDB bei Wärmeabfuhr durch Verdampfung und Einspeisung von 30 °C - Wasser als Funktion der Zeit nach RESA und der in den letzten 2 Tagen gefahrenen Last (zuvor ist Vollast angenommen).

Die Angaben gelten für einen RDB - Druck von 5 bar.

<table>
<thead>
<tr>
<th>Zeit nach RESA</th>
<th>Einspeisemassenstrom bei Ausgangsleistung vor RESA</th>
</tr>
</thead>
<tbody>
<tr>
<td>5 min</td>
<td>15,3 38,0</td>
</tr>
<tr>
<td>10 min</td>
<td>13,4 33,6</td>
</tr>
<tr>
<td>30 min</td>
<td>10,6 26,0</td>
</tr>
<tr>
<td>1 h</td>
<td>8,9 20,9</td>
</tr>
<tr>
<td>2 h</td>
<td>7,5 16,9</td>
</tr>
<tr>
<td>3 h</td>
<td>6,8 15,5</td>
</tr>
<tr>
<td>4 h</td>
<td>6,4 14,2</td>
</tr>
<tr>
<td>5 h</td>
<td>6,1 13,4</td>
</tr>
<tr>
<td>10 h</td>
<td>5,4 11,4</td>
</tr>
<tr>
<td>20 h</td>
<td>4,5  9,5</td>
</tr>
<tr>
<td>1 d</td>
<td>4,4  9,1</td>
</tr>
<tr>
<td>2 d</td>
<td>3,8  7,3</td>
</tr>
<tr>
<td>3 d</td>
<td>3,4  6,3</td>
</tr>
<tr>
<td>4 d</td>
<td>3,1  5,7</td>
</tr>
<tr>
<td>5 d</td>
<td>2,9  5,1</td>
</tr>
<tr>
<td>10 d</td>
<td>2,2  3,8</td>
</tr>
<tr>
<td>20 d</td>
<td>1,7  2,8</td>
</tr>
<tr>
<td>30 d</td>
<td>1,4  2,2</td>
</tr>
</tbody>
</table>

Quellen:  
- Nachwärmeberechnung: Seibold, Abt. UP v. 6.6.88, Basis DIN 25463  
- Einspeisemassenstrom: Stadelmann / Retlinger v. 2.6.88
Schutzziel: Kernkühlung
Maßnahmemodul: Füllstandshaltung RDB F

Erzielbarer Einspeisemassenstrom als Funktion des RDB Druckes

HD / MD - Systeme: (F50 / F60)

1: 1 Reaktorspeispumpe RL BHB
2: 1 TH · HD - Pumpe mit Vorpumpe BHB
3: 1 TH · HD - Pumpe ohne Vorpumpe und 
ohne Primärfüllpumpe F56
(Kavitationsgrenzen TH14 bei p_KK = 0 bar)
4: 2 Vergiftungspumpen F55
5: 3 Dichtungspernwasserumpen TE F54
(Regelventile 100%)
6: 2 Steuerstabantreibspumpen YT F53
(mit HS-Mot. beide Regelventile 100%)
7: Σ (2YT + 3 TE + 2 TW)
8: 1 Kondensatpumpe über RL F61

Temp.Koka
90 °C
keine
Kavitation
80 °C

ND - Systeme: (F70 außer F71)

9: 1 TH · ND - Pumpe BHB
10: 1 Primärfüllpumpe TH F78
11: Feuerlöschspumpen mit der 
mobilen Pumpe TS 8/8S od. TS 16/8-2 über TH20 F75

12: Feuerlöschspumpen LU über RL im Maschinenhaus F72
13: Einspeisung RDB mit der KondiKamer-
entleerungspumpe TM04 D201 über 
TG02 / TH10 F74
14: Nebenkühlwassereinspeisung 
VE20 über TH20
Schutzziel: Kernkühlung
Maßnahmemodul: Füllstandshaltung RDB

Station - black - out
Entleerung SPWB → RDB
Ausgangsleistung: 100%

Reaktorfüllstand [m]

LT3

YC05 L092

2.5 4 6 8 10 12 14 16

23

OK. Kern

Verblockung Umleitstation

LH3 → DDA-RA

LT3 → LADE

LT3 → +200s → SCHADE

Einspeisung SPWB → RDB

Störfallablauf:
Notstromfall mit Ausfall aller Dieselgeneratoren.
Auslaufen des Speisewasserbehälters in den RDB.
Überlaufen von Kondkammerwasser in den Steuerstabantriebsraum
Nach 8h Wiedereinschalten der Kondkammerkühlung. (Annahme).
Ausscheidungen des RDB bis unter Kernunterkante über die Entlastungsleitungen
in die Kondkammer. Danach Aufheizung des Kerns.

Ausfall HWS durch TUSA + Verbl. ULST
RESA
Ansprechen der S/E - Ventile in Funktion Druckbegrenzung
LADE (LT3) 2min 23s
SCHADE (LT3 + 200s) 5min 43s
Einspeisung von Wasser aus dem Speisewasserbehälter in den RDB 8min 20s
Speisewasserbehälter entleert 50min
Füllstand RDB unterhalb OK. Kern 2h 26min
Zr - H₂O-Reaktion 3h
Beginn Schmelzen Steuerstäbe 3h 11min
Absturz des Kerns in das Restwasser im RDB 4h 22min
RDB ausgetrocknet 5h 36min
RDB - Versagen 8h 23min
Entspeichern der Restwärme des abgestürzten Kerns im Steuerstabantriebsraum

Inzwischen durch Überlauf von Wasser aus der KK geflutet
Survey

Part 0 Contents

1. Table of contents
2. Writers guide

Part 1 Introduction and application

1. Introduction into the Emergency Manual
2. Application of Emergency Actions

Part 2 Reactor shut down from Emergency Control Room (TEST)

1.1 Access to TEST 2
1.2 Reactor shut down from TEST 2
2.1 Access to TEST 3
2.2 Reactor shut down from TEST 3

Part 3 Emergency Actions (AiNM)

1. Poisoning reactor coolant
2. RPV - level control
3. RPV - process suppression
4. Heat transfer to environment
5. Suppression pool level control
6. Securing containment isolation
7. Containment venting
8. Coping with H₂ in containment
9. Limiting activity release
10. Securing power supply
11. Securing water supply
12. Containment spraying and flooding
13. Ventilation of the main control room
14. Fuel storage pool cooling
15. Sampling from RPV and containment

Part 4 On site operating of components
Abteilung: PZ  
Bearbeiter: Pappe  
Erstellungsdatum: 01.12.90

NOTFALLHANDBUCH

Teil: 3
Kapitel: 2.15

Einspeisung RDB mit mobiler Feuerlöschpumpe durch Anschluß an VE20 im Kühlwasserpumpenhaus über TH20

Kat. A (Schichtleiterentscheidung)

F 75

Zugehörige Systemschaltpläne: 203-V412-20-1491 VE20, VL20  
203-R311 F-0V-02 TH
<table>
<thead>
<tr>
<th>Contents</th>
<th>Page</th>
</tr>
</thead>
<tbody>
<tr>
<td>Objectives of actions</td>
<td>2</td>
</tr>
<tr>
<td>Prior criteria for actions</td>
<td>2</td>
</tr>
<tr>
<td>Necessary personal- and time requirements for actions</td>
<td>2</td>
</tr>
<tr>
<td>performance of action</td>
<td>3</td>
</tr>
<tr>
<td>Action</td>
<td></td>
</tr>
<tr>
<td>- Description of action</td>
<td>3</td>
</tr>
<tr>
<td>- Action steps</td>
<td>4 - 11</td>
</tr>
<tr>
<td>- Logic for parallel proceeding of action steps</td>
<td>4</td>
</tr>
<tr>
<td>Criteria to terminate the action</td>
<td>12</td>
</tr>
</tbody>
</table>

Appendix A1 title sheet for prepared copies
Appendix A2 Curves
Appendix A3 system drafts
**Ziel der Maßnahme**


**Voraussetzungen zum Durchführen der Maßnahmeschritte**

- mobile Feuerlöscherpumpe TS 8/8S oder ersatzweise TS 16/8-2 betriebsbereit ≤ 5 bar

Bei einem Druck im RDB < 5 bar kann die mobile Feuerlöscherpumpe in den RDB einspeisen, ohne daß das Sicherheitsventil 20VE20S308 anspricht. Eingeleitet wird die Maßnahme bei einem RDB-Druck < 3 bar, um auch ein Ansprechen der Sicherheitsventile im TH-System zu verhindern.

**Personal- und Zeitbedarf zum Durchführen der Maßnahmeschritte**

<table>
<thead>
<tr>
<th>Personalbedarf</th>
<th>Tätigkeitsort</th>
<th>Tätigkeitsblock / Tätigkeit Seite im Kapitel</th>
<th>Zeitbedarf</th>
</tr>
</thead>
<tbody>
<tr>
<td>6 Feuervhleute</td>
<td>Gebäude M5</td>
<td>T1 Feuerlöscherpumpe an VE anschließen Seite 5,6,10,11,12 u. A3 1,2</td>
<td>20 min</td>
</tr>
<tr>
<td>1 Schichtschlosser</td>
<td>RG BO223, BO228, BO123</td>
<td>T2 VE/TH-Systeme Handarmaturen stellen &amp; VE-TH kontrollieren Seite 6, 10, 11, 12 u. A3 1,2</td>
<td>20 min</td>
</tr>
<tr>
<td>1 Schichtschlosser (nur wenn Maßnahmeschritte (11b) u. (12b) notwendig)</td>
<td>RG BO217, BO223, BO612, BO628</td>
<td>T3 Drehmos v. H. v. O. schließen Seite 9</td>
<td>20 min</td>
</tr>
<tr>
<td>1 Schichtschlosser</td>
<td>K4 OK0141</td>
<td>T4 VE-System Dieselstrang absperren Seite 6 u. A3 1, 2</td>
<td>10 min</td>
</tr>
<tr>
<td>1 Reaktorfahrer</td>
<td>Hauptvarte Block B</td>
<td>T5 RS-rücksetzen Drehmos schließen Seite 7, 8, 9, 10, 12</td>
<td>5 min</td>
</tr>
<tr>
<td>1 Schichtelektrik. (nur wenn Maßnahmeschritt (12a) notwendig)</td>
<td>RG BO501</td>
<td>T6 RS unscharf machen Seite 8</td>
<td>10 min</td>
</tr>
</tbody>
</table>

Gesamtzeit bis Maßnahme in Funktion 30 min.
Wirksamkeit, einschränkende Bedingungen

Der erzielbare Einspeisemassenstrom als Funktion des RDB-Drucks ist im Anhang A2 gezeigt. Beispiel: Bei RDB-Druck 3 bar werden 26 kg/s eingespeist.

Maßnahme

a) Beschreibung der Maßnahme (Klartext)


Ist an der Schiene 22 FT Spannung vorhanden, werden gleichzeitig zu den oben erwähnten Tätigkeiten im TH-System die Saugschlieber aus der Koka, die Mindestmengenclieber der Primärfüllpumpe, die Koka-Sprühh- und die Koka-Kühlschieber von der Warte geschlossen.


Ist die Schiene 22 FT spannungslos, wird ein Saugschlieber aus der Koka, ein Mindestmengeschieber der Primärfüllpumpe, ein Koka-Kühl- und ein Koka-Sprühschieber durch Abschrauben des Leistungssteckers unscharf gemacht und von Hand geschlossen.

Achtung

Während der gesamten Maßnahme muß sichergestellt sein, daß ein positives Druckgefälle vom VE-System zum TH-System vorhanden ist. Ein Rückwärtsströmen von Kühlmittel ins VE-System ist unter allen Umständen zu vermeiden.


Ebenfalls sollte darauf geachtet werden, daß der RDB-Druck nicht 3 bar übersteigt, solange die Systemverbindung VE20/TH20 geöffnet ist (siehe Kennlinie Anh. A2), da hier der Bereich des Ansprechens der Sicherheitsventile im TH-System beginnt.

Wichtig

Bei steigendem RDB-Druck ist die Maßnahme bei 11,5 bar Druck RDB zu beenden. Ab hier beginnt die Gefahr der Rückströmung (siehe Kennlinie Anhang A2).

b) Maßnahmeschritte

Logik für paralleles Abarbeiten von Maßnahmeschritten

Schr. = Schritt
T = Tätigkeitsblock

<table>
<thead>
<tr>
<th>T1</th>
<th>Schr. 1 - 6</th>
<th>20 min</th>
</tr>
</thead>
<tbody>
<tr>
<td>T2</td>
<td>Schr. 8, 9</td>
<td>20 min</td>
</tr>
<tr>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>T3</td>
<td>Schr. 7</td>
<td>10 min</td>
</tr>
<tr>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>T4</td>
<td>Schr. 11b, 12b</td>
<td>20 min</td>
</tr>
<tr>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>T5</td>
<td>Schr. 12a</td>
<td>10 min</td>
</tr>
<tr>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>T6</td>
<td>Schr. 13a - 17a</td>
<td>5 min</td>
</tr>
</tbody>
</table>

Mobile Feuerlöscherpumpe speist in den RDB
Diff.-Druck kontrollieren/einspeisen

+ Druck RDB < 3 bar 20 YC 05 P011 ... T5

(18) DRUCK DRUCKSEITE MOBILE FEUERLÖSCHPUMPE
Rückmeldung an Schichtleiter

- DRUCK VE-SYSTEM
  (R-Nr. BO123) KONTROLLIEREN 20 VE 20 P 502 ...
- DRUCK TH-SYSTEM
  (R-Nr. BO123) KONTROLLIEREN 20 TH 20 P504 ...
Rückmeldung an Schichtleiter ...
Druck VE-System ≥ 4 bar Druck TH-System ...
Freigabe Schichtleiter vorhanden ...

- ABSPG NOTEINSPEISUNG VOM VE-SYSTEM
  (R-Nr. BO223) ENTRI/ÖFFNEN 20 TH 20 S106 ...
Rückmeldung an Schichtleiter ...
(mobile Feuerlöscherpumpe speist in den RDB)

Freigabe Schichtleiter vorhanden ...

(19) MIME VERTEILERSTÜCK
Rückmeldung an Schichtleiter ...

Schließen

Kontrolle des möglichen Einspeisemassenstromes in Abhängigkeit des RDB-Druckes (siehe Anhang A2)

+ Füllstand RDB beobachtet 20 YC 05 L011 ... T5
+ Druck RDB beobachtet 20 YC 05 P011 ...
Rückmeldung an Schichtleiter ...

Kriterien für die Beendigung der Maßnahme

Die Maßnahme wird beendet, wenn eine höherwertige Einspeisung zur Verfügung steht.

Sie kann beendet oder unterbrochen werden, wenn der Füllstand im RDB ausreichend ist.

Sie muß beendet werden, wenn der RDB-Druck steigt und 11,5 bar erreicht hat. Ab hier beginnt die Gefahr der Rückströmung.

Bei Beendigung der Maßnahme ist durch Drehzahlreduzierung an der Feuerlöscherpumpe darauf zu achten, daß der Druck auf der Druckleitung nicht über 8 bar ansteigt.

| + Füllstand RDB oder | 16 m 20 YC 05 L092 ... | T5 |
| + höherwertige Einspeisemöglichkeit vorhanden | ... |
| Rückmeldung an Schichtleiter | ... |

Bei zu hoher Strahlung im Raum BO223 wird Schritt (22) Überlaufen.

Freigabe Schichtleiter vorhanden | ...

(22) - NOTEINSP. VON VE ZU TH (R-Nr. BO223)
- ABSP. NOTEINSP. VOM VE-SYSTEM (R-Nr. BO223)
SCHLIEßEN 20 VE 20 S103 ...
SCHLIEßEN 20 TH 20 S106 ...
Rückmeldung an Schichtleiter ...

(23) - ENTLE U. FÜLLANSCHLUSS SCHLIEßEN 20 VE 20 S319 ...
(24) - MOBILE FEUERLÖSCHPUMPE AUSCH ...
Rückmeldung an Schichtleiter ...

Deckblatt Entnahmeexemplar Tätigkeitsblock T2

<table>
<thead>
<tr>
<th>Tätigkeit:</th>
<th>Armatur zu TF schließen, Drücke kontrollieren, Armaturen zu TH öffnen</th>
</tr>
</thead>
<tbody>
<tr>
<td>Tätigkeitsort:</td>
<td>RG R-Nr. B0123, B0223 und B0228</td>
</tr>
<tr>
<td>Personalbedarf:</td>
<td>1 Schichtschlosser</td>
</tr>
<tr>
<td>Zeitbedarf:</td>
<td>20 + 5 min</td>
</tr>
</tbody>
</table>

Das Entnahmeexemplar besteht aus dem kompletten Kapitel 2.15.

Erforderliche Hilfsmittel:

1) Sperrbereichsschlüssel US7  
2) 4 TMI-Schlüssel B2  
3) Dosisleistungsmeßgerät

Aufbewahrungsort:

1) Wartentisch/Schlüsselkasten  
2) Wartentisch/Schlüsselkasten  
3) Notfallschrank 2E Eingang Warte oder Kontrollbereichseingang

Hinweise zur Durchführung der Maßnahmeschritte:

Die Verbindung von VE zu TB darf erst dann geöffnet werden, wenn ein positives Druckgefälle von VE zu TB vorhanden ist und vom Schichtleiter die Freigabe erteilt wurde.

Aufenthaltsort so wählen, daß jederzeit von der Warte Verbindung aufgenommen werden kann.
Einspeisung in den RDB mit der mobilen Feuerlösfpumpe TS16/8-2

- Pumpe steht am Kanalrand und saugt über A-Schlauch
- Anschluß der Pumpe über zwei B-Schläuche an den VE20-Entleerungsstutzen im M5
- Einspeisung über Querverbindung VE20/TH20 in den RDB

Einspeisemassenstrom als Funktion des Reaktordrucks
bei RDB-Füllstand ≤ Speisewasserverteiler und Pegel Einlaufkanal = 429,95 m

VE - Dieselstrang geschlossen (20VE21 S101 ZU)
VE - Hauptstrang geschlossen (20VE20, S104 ZU)

Quelle: KGB Arbeitsbericht PZ 52/88
Fig. 3/11 Example of Emergency Manual procedure
Process and instrumentation diagram (PID)
<table>
<thead>
<tr>
<th>Code</th>
<th>Emergency Action</th>
<th>Achievable performance</th>
<th>required time [min]</th>
<th>personal</th>
</tr>
</thead>
<tbody>
<tr>
<td>F71</td>
<td>Steam pressure main feedwater tank (tank will be emptied) Action will be initiated automatically even at the accident &quot;Station Black Out&quot;</td>
<td>140 kg/s at 1 bar RPV</td>
<td>0</td>
<td>0</td>
</tr>
<tr>
<td></td>
<td></td>
<td>(automat.)</td>
<td></td>
<td></td>
</tr>
<tr>
<td>F72</td>
<td>Suppression pool drain pump by RHR system TH10</td>
<td>36 kg/s at 3 bar RPV</td>
<td>45</td>
<td>3</td>
</tr>
<tr>
<td>F74</td>
<td>Nuclear Service cooling water system VE20 via the connection to the RHR system TH20. Connexion see F74.</td>
<td>70 kg/s at 0.8 bar RPV</td>
<td>30</td>
<td>4</td>
</tr>
<tr>
<td>F75</td>
<td>Mobile fire protection pump by connexion to nuclear service cooling water system VE20 via the connexion to the RHR system TH20. Connexion see F74.</td>
<td>25 kg/s at 3 bar RPV</td>
<td>30</td>
<td>7</td>
</tr>
<tr>
<td>F78</td>
<td>RHR pressure sustain pumps. All pathes except RPV must be shut.</td>
<td>3 x 25 kg/s at 2 bar RPV</td>
<td>5</td>
<td>1</td>
</tr>
<tr>
<td></td>
<td>Safety Objective &quot;Heat sink&quot;</td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>K61</td>
<td>Heat removal from RPV via the reactor water clean up system by encreased operation of both trains.</td>
<td>6 MW at 533 K temperature of coolant</td>
<td>10</td>
<td>1</td>
</tr>
<tr>
<td>V11</td>
<td>Filtered Containment Venting for the purpose of heat removal of the suppression pool (only aloud if the reactor core is still integer) will be initiated above containment pressure ≥ 3 bar.</td>
<td>app. 18 MW at 3 bar cont. pressure</td>
<td>70</td>
<td>9</td>
</tr>
<tr>
<td>Part of Operational Manual</td>
<td>Additional Heat Removal System (ZUNA) 4th redundancy)</td>
<td>80 kg/s at 40 bar RPV</td>
<td>0</td>
<td>0</td>
</tr>
<tr>
<td></td>
<td></td>
<td>(automat.)</td>
<td></td>
<td></td>
</tr>
<tr>
<td>Code</td>
<td>Emergency Action</td>
<td>Achievable performance</td>
<td>Required time [min]</td>
<td>Personnel</td>
</tr>
<tr>
<td>------</td>
<td>---------------------------------------------------------------------------------</td>
<td>------------------------</td>
<td>---------------------</td>
<td>-----------</td>
</tr>
<tr>
<td>C80</td>
<td>Safety Objective &quot;Integrity Containment and Activity Barriers&quot;</td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>C90</td>
<td>Ensuring the isolation of the containment penetrations by closing of back up isolation valves or manual actions.</td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>X61</td>
<td>Spraying control rod drive room by fire protection system (Cat. B).</td>
<td>20 kg/s at 6 bar dry well</td>
<td>5</td>
<td>1</td>
</tr>
<tr>
<td>X41</td>
<td>Spraying containment dry well by spray system TH20 and nuclear service cooling pump VE20 via the connexion see F74. (Cat. B).</td>
<td>75 kg/s at 1,5 bar dry well</td>
<td>30</td>
<td>4</td>
</tr>
<tr>
<td>X21</td>
<td>Spraying the wet well (suppression pool air) by the nuclear service cooling pump VE20 via the connexion see F74.</td>
<td>37 kg/s at 2 bar</td>
<td>30</td>
<td>4</td>
</tr>
<tr>
<td>X111</td>
<td>Flooding suppression pool and dry well by the nuclear service cooling water pump VE20 via the connexion see F74. Flooding dry well will be initiated when RPV melt is suggested. (German containments are designed to be flooded up to the RPV flange. This is required by the Reactor Safety Commission. Due to the danger of a water steam detonation the water mass right on the beginning of an accident is limited. Current this Emergency Action is cancelled until investigations shows that the water steam detonation has no considerable effect to the containment.)</td>
<td>730 kg/s at 2 bar dry well</td>
<td>20</td>
<td>3</td>
</tr>
<tr>
<td>Code</td>
<td>Emergency Action</td>
<td>Achievable performance</td>
<td>Required time [min]</td>
<td>Personal</td>
</tr>
<tr>
<td>------</td>
<td>-----------------------------------------------------------------------------------------------------------------------</td>
<td>-----------------------------------------</td>
<td>---------------------</td>
<td>----------</td>
</tr>
<tr>
<td>X112</td>
<td>Suppression pool flooding by the mobile fire protection pump via the connexion see F74.</td>
<td>32 kg/s at 2 bar</td>
<td>30</td>
<td>8</td>
</tr>
<tr>
<td>V11</td>
<td>Containment venting to prevent breakdown in consequence of overpressure (Cat. B). The venting pressure limit is set to 2/3 of the bursting pressure. (KRB II setpoint 6 bar = 87 psi).</td>
<td>The venting steam flow corresponds to 1 % reactor thermal output = 14 kg/s at 6 bar containment pressure.</td>
<td>70</td>
<td>9</td>
</tr>
<tr>
<td>V12</td>
<td>When containment venting (V11) is in operation, control room venting (V12) is coupled.</td>
<td></td>
<td></td>
<td></td>
</tr>
</tbody>
</table>

**Safety Objective "Limiting Activity Release"**

A
Actions to limit activity release contain manipulations in the ventilation system. As described in Operation Manual.

**Safety Objective "Water Supply"**

V24 Fire protection system feeding of the condensate tank. 30 kg/s 35 4

V34 Fire protection system feeding of the poison solution tank. 10 kg/s 20 2

**Safety Objective "Fuel Storage Pool Cooling"**

There are no emergency actions prepared. If refilling of the fuel pool is necessary it will be done by flexible hose from the fire protection system (Cat. B).
<table>
<thead>
<tr>
<th>Code</th>
<th>Emergency Action</th>
<th>Achievable performance</th>
<th>Required time [min]</th>
<th>personal</th>
</tr>
</thead>
<tbody>
<tr>
<td>S60</td>
<td><strong>Safety Objective &quot;Power Supply&quot;</strong>&lt;br&gt;Auxiliary power supply from 110-kV-Switchplant Gundelfingen.</td>
<td>350 MVA = entire auxiliary power</td>
<td>40</td>
<td>2</td>
</tr>
<tr>
<td>S70</td>
<td>Auxiliary power supply from neighbour unit via switchplant Gundelfingen.</td>
<td>optional</td>
<td>45</td>
<td>4</td>
</tr>
<tr>
<td>S80</td>
<td>Diesels busbars coupling to use any of the 10 diesels for feeding any busbar in each unit (A, B, C). The couplings where backfitted. There are many combinations possible, thus we got 45 instructions per unit.</td>
<td>4,6 MW = nearly the entire emergency</td>
<td>45</td>
<td>1</td>
</tr>
<tr>
<td>S90</td>
<td>Supply of any emergency busbar via underground cable from 20 kV Switchplant LEW Offingen.</td>
<td>6 MW</td>
<td>45</td>
<td>1</td>
</tr>
</tbody>
</table>
Connexion between nuclear service cooling water pump VE20 and RPV injection/residual heat removal system TH20
Fig. 4/9  Backfilting KRB II  
Independed injection-/Heat Removal System (ZUNA)
Fig. 5.1 Practice of Emergency Actions:
Installation of a mobile fire pump for core cooling.
To be seen: The suction nozzle.
Installation of a mobile fire pump for core cooling.
To be seen: The mobile fire pump
Installation of a mobile fire pump for core cooling. To be seen: Connection to the service water by fire hose line.
Installation of a mobile fire pump for core cooling. To be seen: Fix installed fire hose line to connect the pump to the service water system.
Installation of a mobile fire pump for core cooling. To be seen: Connection of the fire hose line to the service water system.
Fig.5/6  Practice of Emergency Actions:
Starting of emergency Diesel by hand on site.
Override of the Reactor Protection System by putting special bridges into the control cubicle.
Override of the Reactor Protection System by insertion of a special control device into the switchgear cubicle.
Refill of the tank of the poison system by fire hose connections
Manually direct switch of 10kV switch on site.
(Override of the control system)
Severe Accident Management

Implementation Activities at Duquesne Light’s Beaver Valley Power Station

Duquesne Light

Roy Brosi
Plant Design

- Single site with two units
- 2652 MWth PWRs
- Westinghouse NSSS and turbine generators sets
- Subatmospheric Containment with quench and recirc spray as ultimate heat removal
Implementation Taskforce

- Project led by the Emergency Preparedness Department

- Taskforce includes:
  - Operations (Procedures)
  - Nuclear Engineering (PRA)
  - Nuclear Training
  - Emergency Preparedness
Implementation Activities

- Guidance development in progress utilizing WOG Severe Accident Guidelines
- The SAMGs will include Control Room and TSC guidelines
- Emergency Plan and Implementing Procedures will be updated to accommodate the SAMGs
# Implementation Schedule

## Severe Accident Management Implementation Milestones

**Beaver Valley Power Station Units 1 and 2**

<table>
<thead>
<tr>
<th>#</th>
<th>Item</th>
<th>Individual</th>
<th>Start</th>
<th>Stop</th>
</tr>
</thead>
<tbody>
<tr>
<td>1</td>
<td>EMS for setps &amp; Calc Aids</td>
<td>Flynn</td>
<td>1/95</td>
<td>2/95</td>
</tr>
<tr>
<td>2</td>
<td>Develop setps &amp; Calc Aids</td>
<td>Frederick</td>
<td>2/95</td>
<td>10/95</td>
</tr>
<tr>
<td>3</td>
<td>Develop SAMGs</td>
<td>Flynn</td>
<td>2/95</td>
<td>9/96</td>
</tr>
<tr>
<td>4</td>
<td>Receive WOG Trng Material</td>
<td>Crum</td>
<td>5/95</td>
<td></td>
</tr>
<tr>
<td>5</td>
<td>Unit 1 IPEEE</td>
<td>Frederick</td>
<td>6/95</td>
<td></td>
</tr>
<tr>
<td>6</td>
<td>Develop BVPS Trng Material</td>
<td>Crum</td>
<td>5/95</td>
<td>4/97</td>
</tr>
<tr>
<td>7</td>
<td>Revise EP Plan/IP</td>
<td>Brosi</td>
<td>4/96</td>
<td>1/97</td>
</tr>
<tr>
<td>8</td>
<td>Perform Trng</td>
<td>Crum</td>
<td>2/97</td>
<td>7/97</td>
</tr>
<tr>
<td>9</td>
<td>Develop Scenarios</td>
<td>Brosi</td>
<td>7/96</td>
<td>7/97</td>
</tr>
<tr>
<td>10</td>
<td>Self Assmnt Drills</td>
<td>Brosi</td>
<td>7/97</td>
<td>8/97</td>
</tr>
<tr>
<td>1</td>
<td>Unit 2 IPEEE</td>
<td>Frederick</td>
<td>7/97</td>
<td></td>
</tr>
<tr>
<td>1</td>
<td>Corrective Actions</td>
<td>All</td>
<td>8/97</td>
<td>10/97</td>
</tr>
<tr>
<td>1</td>
<td>EP Exercise (no degradation)</td>
<td>Brosi</td>
<td>11/97</td>
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<tr>
<td>1</td>
<td>Formal Implementation</td>
<td>Brosi</td>
<td>12/97</td>
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</table>

<table>
<thead>
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<th>Year</th>
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<th>1997</th>
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<tr>
<td>3</td>
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</tr>
<tr>
<td>10</td>
<td>19</td>
<td>20</td>
<td>10</td>
</tr>
</tbody>
</table>

---

Duquesne Light Company
SAMG Responsibilities

- Control Room until TSC available
  - Specific guidance for about 2 hrs
- Technical Support Center
  - Small SAM Team (3 individuals)
  - Reports to the Emergency Director
  - Emergency Director is responsible for decision making
TSC SAM Team

- Expected to include three individuals
- Trained at highest level and able to make recommendations to the ED
- Will utilize existing TSC information (computer data and dedicated lines)
- Decision Flow Chart and precalculate Operational Aids will be used
Termination Guidance will specify safe
Reference to SAMGS and responsibilities
Implementing Procedures will
SAMGS as an available tool
Accident Assessment will include
modified to specify use of SAMGS
Organization and Responsibilities

Emergency Plan Changes
E-Plan Changes (Con't)

- Drills and Exercises will include the general use of SAM drills
- Emergency Response Training will accommodate the addition of SAM Training
- Maintaining Emergency Preparedness will accommodate training and drills
(additional self study expected)

- Implementor 8 hrs
- Evaluator 24 hrs
- Decision Maker 16 hrs

Responsiveness within the ERO

Amount of training depends on type of

EP Training and Operations Training

Responsibilities will be divided between

Training Activities
Implementation Self Evaluation

- Initial (upon creation)
  - Ensure feasibility and usefulness
  - Integration without degradation

- Ongoing (subsequent)
  - Table-top and interfacility mini-drills
  - Periodic (not more than 6 yrs) with full critique and feedback
  - Train, evaluate, and improve
Summary
OECD SPECIALIST MEETING ON
SEVERE ACCIDENT MANAGEMENT IMPLEMENTATION

Niantic, Connectiicut, USA, 12th - 14th June 1995

Provisions for Emergency Situations
in the Krümmel Nuclear Power Plant

W. Stubbe
U. Welte

Kernkraftwerk Krümmel GmbH
21502 Geesthacht
Deutschland
## CONTENTS

Abstract

1. Introduction

2. Emergency response organisation and administration

3. Regulations and procedures

4. Technical measures

5. Emergency training programme

6. Involvement of the regulatory authority
ABSTRACT

Work on emergency preparedness started at the Krümmel nuclear power plant before the first start up in September, 1983; an emergency response plan had been created and an emergency response team had been formed. According to our experience gained in exercises the emergency planning had been improved.

During the last eight years some additional emergency response actions, so-called accident management measures, had been implemented.

The KKK emergency response organisation and administration ensure an effective planning, implementation and control of all emergency preparedness activities. Up to the present the implementation of new accident management measures didn't require changes of the organisational provisions.

Regulations and procedures which are needed to cope with events of any type are put together in a special part of the operational manual and in the emergency manual. The operational manual provides procedures for incidents and design basis accidents, the emergency response measures described in the emergency manual have to be performed only in the range beyond the design boundaries.

In order to ensure an effective and appropriate application of the prepared procedures safety goals had been defined which have to be met. The structure of all procedures is similar. Contents and structure meet the requirements. An incident/accident diagnosis guide helps the operator to choose the appropriate approach to cope with an event. Based on the results of the probabilistic risk analysis accident management measures had been developed and implemented. Measures for prevention of severe accidents and for mitigation of consequences had been prepared. Number and type of accident management measures are in accordance with the state of the art in Germany.

- 1 -

(0078U)
Despite of some technical problems of minor importance and licensing problems no peculiar difficulties arose in connection with the implementation of the new measures.

In 1985, an emergency training programme had been established at KKK, therefore a new special training programme wasn't needed after the implementation of the new accident management measures.

Two additional accident management measures will be implemented in the near future. KKK intends to install an emergency activity control instrumentation and a new containment sampling system. Other necessities of improvement or implementation of new measures may arise in the future depending on the progress in research on severe accidents.

1 Introduction

The 1300 MW nuclear power plant Krümmel (KKK), a BWR (type 69) constructed by Siemens/KWU, is located on the banks of the Elbe river in the northern part of Germany close to the city of Hamburg.

The plant went into operation in september 1983, therefore the main lessons learned from the core melt accident at Three Mile Island-2 could be taken into account while the plant was still under construction.

Work on emergency preparedness started in KKK before the first start up of the plant. First, an emergency response plan had been created and an emergency response team had been formed.

According to the experience gained in on-site emergency exercises, supporting documents, special equipment and procedures had been prepared and optimized.
After the accident at the Chernobyl nuclear power plant in 1986, our work focussed on the implementation of additional emergency response actions, so-called accident management measures.

Accident management measures had been prepared for the prevention of severe accidents and for the mitigation of the consequences.

Emergency management at the Krümmel nuclear power plant consists of the following components:

- An emergency response plan.
- Emergency response organisation and administration.
- Emergency facilities, equipment and resources.
- Technical support by the emergency response organisation of the plant constructor Siemens/KWU.
- Emergency assessment and notification procedures.
- Emergency response actions for protection of the reactor core and of safety-related systems and actions for minimizing the activity releases.
- An emergency training programme.

We have to ensure that these provisions will be available and effective at any time. On this account, work on emergency preparedness is a "never ending" work and there is still room for improvement.
Emergency response organisation and administration

The KKK emergency response organisation and administration ensure an effective planning, implementation and control of all emergency preparedness activities.

The organisational structure is clearly defined; the structure of the team is shown at picture 1.

![Emergency Response Team of the Krümmel Plant](image)

The tasks which the team members have to take over can be divided into two types:

- Tasks which are similar to work during normal operation of the plant.
- Special tasks which only have to take over during an emergency situation.

The main tasks are:

- Assessment of plant status and of the radiological situation.
- Classification of event and notification.
Planning and performing measures in order to meet the safety goals and to prevent a severe accident and to mitigate the consequences of the accident.

Providing timely and accurate information to organisations and people with a need to be informed.

Ensuring an effective communication and documentation.

Realising the best support to the personnel which operates and controls the systems of the plant.

Supply the staff with all needed materials and services.

The responsibilities of the groups are shown at /picture 2/.

**Responsibilities of the Emergency Response Team**

<table>
<thead>
<tr>
<th>Team Leader</th>
<th>Shift Division</th>
<th>Support Group</th>
<th>Mechanical Engineering</th>
<th>Electrical Engineering</th>
<th>Radiological Protection</th>
<th>Communication</th>
<th>Information</th>
<th>Supply</th>
</tr>
</thead>
<tbody>
<tr>
<td>Diagnosing, Assessment of plant status and the radiological situation</td>
<td>Performing of Emergency actions</td>
<td>Performing of Repairs and Emergency actions</td>
<td>Radiological Surveillance</td>
<td>Health Physics</td>
<td>Industrial Safety and Plant security</td>
<td>Information of the Emergency Response Team of the government</td>
<td>Information of the public</td>
<td>Supply with materials and services</td>
</tr>
</tbody>
</table>

The leading of the team is taken over by the plant manager or one of his substitutes.
Criteria for the activation of the emergency response team are described in the "alert-chapter" of the operational manual.

Within less than one hour after notification the team is ready for work. The team is working at the emergency response center which is located close to the control room.

3 Regulations and procedures

Regulations and procedures which are needed to cope with events of any type (incidents, design basis accidents and severe accidents beyond the design basis) have been put together in a special part of the operational manual and in the emergency manual.

The regulations and procedures of the operational manual are applicable to incidents and design basis accidents, the emergency response measures described in the emergency manual only have to be performed in the range beyond the design boundaries.

The operational manual provides two different types of procedures:

- Incident-related procedures and
- symptom-based procedures.

The emergency manual consists of three parts:

Part I: (a) Organisation and administration

  e. g.  * Structure of the emergency response team
         * Tasks and responsibilities of the team
         * Special regulations with regard to

- 6 -
e. g.
- Alert classification
- Activation of the team
- Assessment of plant status and situation in the environment
- Decision-making
- Communication
- Documentation

(b) Emergency training programme.

Part II: Working manual of the team which consists of supporting documents, records, data and other material and informations needed by the groups of the team

  e. g. * Check lists
         * Registration forms
         * Measuring programmes
         * Plant status reports
         * Calculation methods.

Part III: Procedures for accident management measures

(Symptom based procedures).
In order to ensure an effective and appropriate application of the prepared procedures safety goals had been defined which have to be met.

The safety goals of the Krümmel plant are shown in /table 1/.

<table>
<thead>
<tr>
<th>Safety Goal</th>
<th>Parameter</th>
<th>Criterion</th>
</tr>
</thead>
<tbody>
<tr>
<td>1. Subcriticality</td>
<td>neutron flux</td>
<td>&lt; 1000 imp/s</td>
</tr>
<tr>
<td>2. Core Cooling</td>
<td>Coolant level in</td>
<td></td>
</tr>
<tr>
<td></td>
<td>the reactor</td>
<td></td>
</tr>
<tr>
<td></td>
<td>pressure vessel</td>
<td></td>
</tr>
<tr>
<td>3. Core Cooling</td>
<td>Reactor pressure</td>
<td>&lt; 74 bar</td>
</tr>
<tr>
<td>Pressure Control</td>
<td></td>
<td></td>
</tr>
<tr>
<td>4. Core Cooling</td>
<td>Water temperature</td>
<td>&lt; 55°C</td>
</tr>
<tr>
<td>Heat removal</td>
<td>Suppression pool</td>
<td></td>
</tr>
<tr>
<td>5. Core Cooling</td>
<td>Containment pressure</td>
<td>&lt; 0.25 bar</td>
</tr>
<tr>
<td>Pressure reduction</td>
<td></td>
<td></td>
</tr>
<tr>
<td>System</td>
<td>Water level in the</td>
<td>&gt; 20 m</td>
</tr>
<tr>
<td></td>
<td>Suppression pool</td>
<td></td>
</tr>
<tr>
<td>6. Control of</td>
<td>Activity release rate</td>
<td></td>
</tr>
<tr>
<td>Activity release</td>
<td>radionuclides-131 &lt; 3E+67 Bq/m</td>
<td></td>
</tr>
<tr>
<td></td>
<td>noble gas &lt; 7E-12 Bq/m</td>
<td></td>
</tr>
</tbody>
</table>

/table 1/

In the case of an unusual event the operator has to decide which approach to cope with the incident has to be chosen.
For this purpose the operational manual provides an "Incident/Accident Diagnosis Guide" which is schematically described at /picture 3/.

Incident/Accident Diagnosis Guide of the Operational Manual

Incident
→ Automatic actions
→ Check of Safety systems function and Check Safety Goals.
→ Safety goals met?
→ YES

Identification of Incident type
→ Incident type identified?
→ YES

Initiation of incident-related actions
→ Safety goals met?
→ YES
→ NO
→ End

Initiation of symptom-based actions (operational manual)
→ Safety goals met?
→ YES
→ End

Initiation of symptom-based actions (Emergency manual)

-picture 3-
Technical measures

Accidents beyond the design base were investigated in the frame of probabilistic risk analysis (PRA). Results of the studies show that there are large built-in safety margins in the plants which give the plant personnel the opportunity to take additional manual actions to prevent severe accidents or to mitigate the consequences of them.

On the base of the results of the PRA accident management measures had been developed. They rely on the flexible use of the plant systems and the use of some mobile equipment. For some measures it was necessary to install new systems or to backfit components or systems.

There are two categories of accident management measures: Measures for prevention and measures for mitigation.

The most important measures, implemented in the Krümmel plant are described below:

(1) Installation of an emergency reactor pressure control system

At KKK, the control of the reactor pressure is ensured by eleven safety valves of a similar type. Although the probability of independent failures affecting all valves at the same time is extremely low there is a risk for common mode failures which can not be quantified. This risk was eliminated by the installation of additional valves of a different type.

(2) Installation of a filtered containment venting system

To ensure the integrity of the containment a pressure relief system had been installed which will be activated if the pressure in the containment reaches the design limit. The system is provided with Venturi scrubbers and a wire mesh filter system.
(3) Installation of a containment nitrogen-injection system

In order to prevent the formation of an explosive gas mixture the containment is inerted with nitrogen. In the case of an emergency the injection of nitrogen takes too much time, therefore the containment is inerted even during normal operation.

(4) Improvement of the steam-driven high pressure emergency cooling system

The auxiliary drives of this system had been connected to the battery-supplied alternating voltage.

Now it is possible to operate the system for about four hours in the case of a total failure of the emergency power supply system.

(5) Increasing the capacity of the batteries

(6) Installation of a filtered supply air system for the control room

(7) Improvement of the emergency power supply system by connection the system with another 110 kV line and a hydro power station (pump-fed power plant).

(8) Implementation of several additional measures for the injections of water into the reactor pressure vessel:

   * Improved operation of the steam driven high pressure emergency cooling system.

   - 11 -
** Use of the contents of the feedwater tank.
** Injection of water by the control rod cooling water system.
** Use of the seal water system.
** Injection of coolant by the demineralized fire-fighting system.
** Injection of riverwater by mobile fire-fighting pumps.
** Coolant supply by the demineralized water storage tanks.
** Use of the local drinking water supply system.
** Injection of coolant by means of the fuel pool cooling system.

The several injection opportunities are shown at /picture 4/.

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**Possibilities of water injection at the KRONHEL plant**

- 12 -
Other measures or improvements had been implemented, they are of minor importance.

** PROCEDURES **

According to the german design philosophy the need of operator actions which are necessary to perform the prepared measures or to operate the new installed systems had been minimized.

The procedures needed are comprized in Part III of the emergency manual.

They had been structured in the same way like the procedures of the operational manual.

**Structure of emergency procedures**

(1) General instructions
   ** Aim of the measure
   ** Description
   ** Criteria for the initiation and ending of the measure
   ** Requirements of conditions to perform the measure
   ** General instructions and advices.

(2) System drawings

(3) Equipment needed

(4) Personnel requirements

- 13 -
(5) Detailed description of the actions  
(including the efficiency aimed at)

(6) Possibilities to control the efficiency

(7) Plans of rooms  
(photos of rooms and equipment)

(8) Diagrams

(9) Copies of the procedure  
(to take along with the operator).

- PROBLEMS

Despite of normal technical problems of minor importance no  
special problems arose in connection with the implementation of  
the accident management measures. There had been no need for an  
 improvement of the Post Accident Instrumentation.

However some licensing problems had to be solved.

- NECESSITY OF CHANGES IN THE ORGANISATION

The implementation of accident management measures after the  
first start up of the plant didn't lead to any change of the  
organisation because an emergency response team already had been  
formed before.

New measures only add to the number of actions the emergency  
response team has to perform for prevention of severe accidents.
TRAINING

The implementation of accident management measures during the last years did not require an additional training programme. The actions which have to be performed in emergency situation have to be very simple (if they are complicated, they are not useful), therefore it is very easy to learn to operate the new systems and to deal with the new procedures.

The necessary training is included in the normal training programme which is required to ensure the qualification of personnel, and in the emergency training programme.

Since the first start up of the plant the people in KKK work on emergency preparedness, they have to absolve emergency training measures periodically, every member of the emergency response team is used to discussions on "accidents beyond the design basis."

Therefore emergency management is a part of work during normal operation, the implementation of new accident management measures is "all day work."

PLANNED MEASURES

Two additional accident management measures are in the planning stage.

(1) The activity-control-instrumentation is not sufficient to control the activity release to the environment in a case of a severe accident. Therefore we intend to install a special emergency instrumentation. The construction work will be completed soon.
(2) The system, which is available for taking samples from the containment atmosphere, in order to assess the state of the core and to assess the activity release potential also do not meet our requirements.

The installation of a more sufficient equipment is necessary.

Other necessities of improvement or of implementation of new accident management measures may arise in the future depending on the progress in research on severe accidents.

5 Emergency training programme

The training programme is an essential part of the provisions for severe accidents. Emergency management can be effective and successful only, if the members of the emergency response team have absolved a sufficient training.

The training programme consists of the parts listed below:

- Training of parts of the emergency plan
  
  e. g. * Emergency event classification
  * Activation of the emergency response organisation
  * Technical assessment
    (e. g. plant condition, radiological hazards)
  * Information of the authority and the public.
- Training of the divisions of the emergency response team
  
  e. g. * Table top exercises of team leaders
  * Simulator training of operators
  * Teamwork in divisions.

- Emergency exercises with activation of the whole staff.

The above described training programme had been established in 1985. Since then every member of the emergency response team had to take an active part in minimal one arrangement per year.

Especially important are the emergency exercises which take place once or twice per year. This exercises are "Full Field Exercises" which test the entire structure of emergency preparedness established at our plant. The exercise scenarios are developed as realistic as possible; also accident management measures are performed as realistic as the safe plant operation do allow.

Emergency exercises are suitable to review procedures and equipment and to check whether the staff is well qualified and trained or not.

6 Involvement of the regulatory authority

According to german atomic law, the implementation of accident management measures requires a licensing procedure if the plant safety level is affected by the measure.

Therefore for example licenses were needed for the installation and the operation of the filtered containment venting system and the containment nitrogen injection system.
An approval of the licensing authority is necessary in the case of changes of plant systems and equipment and of operating procedures.

But there is no need for a license when provisions for severe accidents are changed without affecting any licensing documents. The emergency manual is not licensed.

A general review of our emergency preparedness by the authority, is periodically performed during our annual emergency exercise.
IMPLEMENTATION OF SEVERE ACCIDENT MANAGEMENT MEASURES ON NUCLEAR ELECTRIC POWER STATIONS

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Nuclear Electric plc.

Abstract

Accident management in the UK is implemented as part of the overall emergency planning arrangements. These are common to all reactor types. The details of the accident management measures are dependent on the reactor type. The paper outlines the Nuclear Electric's overall approach to emergency planning and the way in which procedures for coping with beyond design basis accidents on both GCRs and PWRs have been developed and implemented. A more detailed discussion of the development and implementation of the procedures for the Sizewell 'B' PWR is presented to illustrate the way in which preventive measures and "accident management" measures have been integrated in the development of a modern plant.

1. INTRODUCTION

Accident management in the UK is implemented as part of the overall emergency planning arrangements. These are common to all reactor types. The details of the accident management measures are dependent on the reactor type. In addition the approaches adopted for the gas cooled reactors (GCRs) and for the PWR differ because of differences in the stage of development at which "beyond design basis" considerations were introduced. In the case of the GCRs these had to be added to the existing operating procedures whereas for Sizewell 'B' (the UK's first commercial PWR) such accidents were considered in producing the initial procedures.

In a previous paper (Ref 1) the overall approach used for the development of accident management procedures for Sizewell 'B' was outlined. This paper will provide an update to this as well as outlining the parallel developments on the operating gas cooled reactors.

Section 2 describes the overall emergency planning and organisational aspects. The off-site organisation has changed since reference 1 to bring it in line with that used for other civilian emergencies. Sections 3 and 4 outline the approaches to severe accident management on the gas-cooled and PWR reactors respectively.
OVERALL APPROACH TO EMERGENCY PLANNING

Accident management in the UK is implemented as part of the overall emergency planning arrangements; these are common to all reactor types. Each station is required to define and implement a detailed emergency plan which is formally submitted to the Nuclear Installations Inspectorate for their approval. Traditionally the detailed emergency plan has always been based on a "reference accident" which is normally the design basis accident with the worst off-site radiological release. This detailed plan provides the basis for the emergency arrangements including training and exercises to demonstrate its effectiveness. However, even though accidents more severe than the most limiting design basis faults are highly unlikely, as part of the overall "defence in depth" philosophy, the arrangements supporting the detailed emergency plan form the basis for the extension of the emergency arrangements to cope with more extreme faults. That this extension is feasible has had to be demonstrated as part of the supporting case for the adequacy of the emergency arrangements.

For the older Magnox reactors the most severe accidents considered within the design basis resulted in radiological releases which could require limited off-site evacuation based on the Emergency Reference Levels (ERLs) for evacuation defined by the National Radiological Protection Board (NRPB). The anticipated extent of evacuation for these faults (together with allowances for uncertainties, local conditions etc.) then defined the detailed emergency planning zone. This generally covers an area of 2 to 3 kms around the site. The later Advanced Gas Cooled (AGR) reactors and Sizewell 'B' were designed so that off-site evacuation would not be required for any fault within the design basis. However even in such cases a minimum detailed planning zone (of about 1 km) is required to form the basis for extension of the emergency planning arrangements to cope with the more unlikely beyond design basis situations. In the case of newer reactors sharing a site with an older Magnox station (e.g. Sizewell 'B'), the emergency planning zone of the older station provides the basis for the emergency arrangements for both stations.

The emergency arrangements for a nuclear power station have many features in common with those in place to cope with major non-nuclear civilian emergencies. One feature of the nuclear arrangements not commonly found in other plans (see reference 1) was that, until recently, Nuclear Electric was responsible for the establishment of an off-site support centre at which offsite aspects of the response to an accident could be co-ordinated. This reflected the need for specialist advice to be provided to those taking decisions on the off-site response. The review of UK emergency arrangements carried out following Chernobyl, which also took account of Police experience in co-ordinating the response to a number of recent major civil emergencies, has led to changes which have brought the framework for the off-site
response to a nuclear accident into line with that for other emergencies whilst still providing the required level of specialist support.

The overall, common aspects of organisational responsibilities and interfaces, will be presented before outlining the reactor specific approaches to guidelines and procedures.

2.1 Organisational Responsibilities

On the declaration of either a Site Incident or Nuclear Emergency, the Nuclear Electric emergency organisation is set up. This involves the manning of three emergency centres:

1. ECC - Emergency Control Centre - on site
2. LEC - Local Emergency Centre - off site centre attached to the local Police HQ
3. CESC - Central Emergency Support Centre - at Nuclear Electric's HQ

The station emergency organisation is led by the Emergency Controller who is a trained and authorised member of the station's staff (Station Manager or other senior staff). The Emergency Controller is responsible for all actions taken on site (with the sole exception of the control of fire fighting once the local Fire Service is in attendance). He is based in the site ECC which is provided with all the facilities necessary for it to function in an emergency. (It is generally located in a separate building from the Main Control Room (MCR) but (for the PWR) in the same building as the Technical Support Centre (TSC) and the Auxiliary Shutdown Room (ASR)). The MCR retains its role as the centre from which control of the plant is exercised.

Following the declaration of the Incident or Emergency the Emergency Controller (EC) is responsible for initiating formal notification to off site organisations and in the case of a Nuclear Emergency, for providing advice to those organisations with responsibilities for taking actions to protect the public. This is done via the Police Liaison Officer in the ECC. The EC retains this responsibility until the CESC is operational.

The LEC is managed by the Police with a senior police officer in overall control. The LEC would be available almost immediately and would be fully set up in about 1 hour. In addition to the police, in the event of a Nuclear Emergency, staff from NE, the National Radiological Protection Board (NRPB), the Nuclear Installation Inspectorate (NII), a Senior Government Liaison Representative and a Government Technical Advisor (GTA) would attend. Until the arrival of the GTA, NE provide
the Technical Advisor.

The CESC is established to relieve the affected site of responsibility for providing advice on all aspects of off-site radiological protection and to provide a single source of technical support to the site to assist them in bringing the situation under control. The CESC will also provide expert advice to the Nuclear Electric Technical Advisor and subsequently the Government Technical Advisor at the LEC.

Throughout the emergency a number of organisations will work closely together. These include NE, the police, the local authorities, the Fire service, the NII, the Health Authority, various Government Departments and other parts of the UK Nuclear Industry but it remains a fundamental principle that the Emergency Controller retains ultimate responsibility for on site actions.

Although the emergency controller is the focus for all on-site activities and ultimately controls the MCR staff as well as all the emergency teams, it is normal when considering guidance and procedures to differentiate between the requirements of the MCR operators and the rest of the Emergency Controller's support staff.

In the short term, it is the shift staff who will be responsible for the early actions. Outside normal working hours the Shift Charge Engineer (the shift manager) assumes the role of emergency controller and will be responsible for the initial declaration of the emergency. The MCR staff will need to take the appropriate actions to control the plant in the short term. For all design basis accidents the required actions are included in the Station Operating Instructions (SOIs). As the emergency organisation is activated more and more support becomes available to the Emergency Controller; firstly in the form of on-site engineering staff and later, off-site engineering and specialist technical staff. The provision of consistent guidance to the on-site staff is valuable, firstly because they are available immediately, and secondly because they are not hindered by the lack of communications. Any such technical guidance is also made available to off-site teams to establish a consistent starting point.

The development of beyond design basis and severe accident procedures has taken place in the period following the accidents at TMI-2 and Chernobyl. The overall approach for the Gas Cooled Reactors and Sizewell 'B' was dictated by the differences in technology and the different states of development of the plants. The GCRs were already in operation, and had established operating procedures on which the operators had been trained and were thoroughly familiar. Sizewell 'B', on the other hand was the first commercial PWR to be built in the UK and was still in the detailed design phase at the time of the accident at TMI. This has led to differences in the way in which the procedures have been implemented. In the next section the approach adopted on the GCRs will be outlined.
3. APPROACH TO ACCIDENT MANAGEMENT FOR GAS COOLED REACTORS

In line with best international practice, Nuclear Electric is implementing a comprehensive strategy for dealing with hypothetical accidents which extend beyond the design basis of the plant. The strategy is based on the provision of procedural guidance to reactor operators and other staff with accident management responsibilities. The approach followed sought to minimise the impact on existing fault recovery procedures.

The approach is discussed in more detail in reference 2 but will be outlined here. The spectrum of accidents outside the design basis is divided into two: those accidents which go beyond the design basis but have not progressed to core damage (termed Beyond Design Basis Accidents (BDBA)) and those accidents involving core damage. To cope with this potential spectrum two sets of guidance have been formulated: Symptom Based Emergency Response Guidelines (SBERGs) which are focused on the prevention of core damage, and Severe Accident Guidelines (SAGs) which focus on the mitigation of activity releases to the environment. The procedures are designed to provide a degree of overlap with each other and with the existing event based procedures as is illustrated in figure 1.

3.1 Symptom Based Emergency Response Guidelines

The SBERGs focus on the maintenance of 4 Critical Safety Functions (CSFs). These are: sub-criticality, pressure vessel integrity, cooling and the control of radioactive release. The choice, whilst not unique, corresponds approximately to the timescales for concern in extreme faults. Achievement of adequate neutronic shutdown is the immediate priority following a fault. Over-pressurisation of the primary circuit could be of concern on somewhat longer timescales but deficiencies in post-trip cooling would not generally threaten fuel or plant failures for several hours because of the large thermal inertia of the reactors. The control of radioactive release is only of significance following fuel failure, which is unlikely to occur until one or more of the other CSFs have failed.

Entry into the SBERGs is via a monitoring procedure which is followed at every trip. This identifies whether any CSFs are threatened and directs the operator to the appropriate SBERG. The guidance in the SBERGs relies on best estimate calculations of both transient analysis and structural integrity aspects of the faults. These reveal that, in general, long timescales are available for recovery. This is discussed in more detail in reference 2.

SBERGs have been prepared individually for all AGR and Magnox stations based on plant specific assessments.
3.2 Severe Accident Management Guidelines

Since the power density in GCRs is relatively low and the thermal capacity of the core structures is high, very long timescales (of order of one day or more) exist before there is a risk of serious core or plant damage. Thus there is a high probability that the SBERGs will be successful in avoiding core damage. Nevertheless, the SAGs are provided to give guidance for coping with such unlikely situations. The priority changes from prevention of damage to the core and plant to the mitigation of radioactive release. Due to the uncertain state of the plant, the advice is written as very general guidance which is generic to a particular reactor type rather than in procedural format for individual reactors. Furthermore the advice is directed towards the Emergency Controller since it includes actions for the damage repair teams as well as MCR staff.

The advice is divided into two separate plant states covering pressurised and depressurised faults since the need for and feasibility of specific actions can depend strongly on whether the vessel is pressurised. In general, depressurisation is recommended at a relatively early stage to delay or prevent failure of the primary circuit and to minimise the dispersal of activity.

It is recognised that some SAG advice runs counter to normal operating procedures or must address conflicting requirements. The SAGs therefore address both the advantages and disadvantages of the recommended recovery strategies. To support the SAGs a detailed R&D programme, jointly funded by the UK GCR operators has been undertaken. More details on the specific advice is given in reference 2.

4. APPROACH TO SEVERE ACCIDENT MANAGEMENT FOR SIZEWELL 'B'

Since Sizewell 'B' is the first commercial PWR to be built in the UK and because its detailed design development was undertaken in the period immediately following the TMI-2 accident, the decision was taken to produce a suite of operating procedures to cover all operating states from normal operation to severe accidents. This decision predated the work on GCRs discussed above so there was no UK precedent to follow and the difficulties associated with grafting on new procedures to existing ones did not arise.

The Sizewell 'B' Station Operating Instructions (SOIs) include symptom based procedures (SOI 8) based on the monitoring of Critical Safety Functions. These are, in normal priority order:

1. Sub-criticality
2. Core Cooling
3. Heat Sink
4. RPV Integrity (PTS)
5. Containment
6. Inventory

Included in SOI 8 are the Severe Accident Mitigation Procedures (SOI 8.8). The structure of SOI 8 and the way in which the severe accident procedures are incorporated was presented at the previous workshop (ref 1). The approach differs from that on the GCRs in that above a certain level of challenge the operators are directed to use the SOI 8 procedures, whereas the advice contained in the SBERGs is not mandatory. The entry and return conditions are obviously easier to incorporate into a new set of procedures than to add to existing ones.

Entry into the severe accident procedures is generally from the core cooling CSF and is initiated by a high core exit temperature measurement. When SOI 8.8 is entered the priorities of the CSFs change from that implied by the listing above to maintenance of containment being the top priority with core cooling continuing to be pursued as a secondary one. The actions which are included in such procedures are largely associated with the use of existing plant in different modes and with different (more relaxed) limits applied. Because severe accidents were considered in the design of Sizewell 'B' it has not proved necessary to provide additional equipment for severe accident management. The development of the procedures will be briefly outlined in the next section.

4.1 Development of the Sizewell 'B' Severe Accident Mitigation Procedures

The previous paper (ref. 1) outlined the early development of the procedures. Here this will be summarised and subsequent developments outlined.

As was noted above, the intention was that the SOIs would be developed to cover all operational conditions. The Sizewell 'B' design set out to provide engineered protection for all fault sequences with frequencies of $10^{-7}$ or greater. However there was an additional requirement to extend the analysis beyond the design base envelope to ensure that there was no sudden disproportionate increase in risk immediately beyond the design base limit. This was done by carrying out PSAs for internal initiators at power at an early stage in the design development (in support of the Pre-Construction Safety Report). These PSAs confirmed the overall adequacy of the design but highlighted a number of improvements which could be made (ref 3). In addition to the analysis which focused on the level 1 results, a level 3 PSA had been included. This gave insights into a region which was technically "beyond the design basis". Nevertheless a number of design changes and design refinements were made to further reduce the severe accident risk, in line with the ALARP principle. The main ones were:

i) provision of additional isolation valves on the RHR suction lines to reduce the interfacing system LOCA (V sequence) probability.
increasing the capacity of the diverse heat rejection route from the containment fan coolers to the Reserve Ultimate Heat Sink. (This was included as a design provision for certain refuelling faults but by ensuring that it had the capacity to cope with severe accidents significantly improved the containment system reliability.)

alteration of the containment floor levels and kerb heights and the inclusion of a passive “flap-valve” in the instrument tunnel to ensure that for all severe accidents there would be between 1 and 3m of water in the reactor cavity prior to RPV failure, even without the operation of engineered safeguards. This water would then be available to quench core debris following containment failure. In arriving at this solution it was necessary to balance the benefits of debris cooling with the potential disbenefits of an ex-vessel steam explosion. The latter was not considered to be a threat to containment integrity because the size of the Sizewell 'B' containment is such that direct pressurisation is not a problem and the cavity geometry is such that the creation of a damaging missile is unlikely. At the time that the decision was made it was considered that the maintenance of a coolable geometry was likely to be more effective with an initially quenched debris bed rather than trying to quench a molten mass by pouring water onto it. Steam explosions could have a detrimental effect on coolability in that the material involved in the steam explosion will be more finely divided. However it is also likely to be more widely dispersed, leading to shallower debris beds. In addition the presence of water in such quantities is likely to have a mitigative effect on direct containment heating following high pressure melt ejection. Thus on balance having water in the cavity prior to RPV failure was considered preferable.

In addition to having a number of direct impacts on the design, the PSA results also helped to guide some of the decisions as to which potential accident management measures should be included. A number of these had already been included as a result of deterministic considerations. The main addition made to SOI 8.8 was the inclusion of deliberate depressurisation. Although the indications from preliminary analysis and experiments were that Sizewell 'B’s containment would survive a high pressure melt ejection, it would reduce the uncertainties if the primary circuit was depressurised. Since both PORV’s and one code Safety Valve on Sizewell 'B’ have been replaced by three pairs of SEBIM POSRVs, the vent capacity available under operator control is relatively large. It should be noted that in many sequences depressurisation is invoked at an early stage to make use of low pressure water sources. The procedure included in SOI 8.8 allowed depressurisation under severe accident conditions irrespective of the availability of water sources.

One issue which was considered at this stage was the provision of a containment
filtered venting system. On the basis of the preliminary PSA this was not considered to be justified since such features as the diversification of the fan cooler heat sink had resulted in a sufficiently high heat removal reliability that the predicted frequency of late over-pressure failures was relatively low. The provision of an additional system could not be justified on cost benefit grounds and the inclusion of an additional system which potentially compromised the containment function, given the assessed high reliability of the containment systems, was not considered to be beneficial. However, since the decision rested on a probabilistic argument it was agreed that one of the containment penetrations would be qualified for operation with a filtered venting system to allow one to be back-fitted, if necessary.

As part of the Pre-Operational Safety Report a full level 3 PSA was carried out covering all initiators at all power states. In addition to providing further confirmation of the design adequacy it was envisaged that it would be used to help refine operational procedures and Technical Specifications. To support the level 2 aspects of the analysis an extensive R&D programme was undertaken. This involved participation in international R&D programmes such as CSARP and ACE as well as work in the UK and addressed the complete range of severe accident issues. Although, initially targeted on supporting the PSA, the studies also provided the basic information required to validate the decisions made with respect to accident management. For instance, an extensive study was carried out on steam explosions, focusing on the "alpha" mode. This covered the probabilities of vessel failure as a result of steam explosions at various different pressures, showing that the alpha mode probability was always low, but did show a peak at intermediate pressures. This information could then be used to review the decisions made about intentional depressurisation. The predicted increase in failure probability at intermediate pressures was not enough to out-weigh the potential benefits of depressurisation particularly when it was noted that by including such actions in procedures, it was likely to be accomplished at an early stage in the core degradation before a steam explosion was likely.

The overall PSA results confirmed the adequacy of the design and procedures for faults at power but did lead to some further refinements, particularly for faults at shutdown. The risk from faults at shutdown was dominated by faults initiated whilst at mid-loop. Under these circumstances the primary circuit inventory is reduced and the design feature which was intended to ensure that there was sufficient water in the cavity to quench the debris was ineffective. Thus basement attack ensued leading to containment failure as a result of basement penetration or overpressurisation due to very late, very large hydrogen burns (ref 4). The base case PSA had been carried without making extensive claims for accident management actions so the possible impact of such actions was examined. Faults at mid-loop can be mitigated by gravity feed from the RWST or by the restoration of safeguards. Sensitivity studies were carried out on late restoration of sprays (ref 4) which showed a significant benefit.
In addition to the normal reactor building spray system, the fire suppression system extends into the containment. The fire sprays are relatively low in the containment but analysis was undertaken to demonstrate that this system will be effective in reducing the reactor building pressure as well as providing water to quench the debris. As a result of this the initiation of the containment fire protection system, after RPV failure, was added to SOI 8.8. The inclusion of this in the analysis also confirmed that a filtered venting system was not required to satisfy the ALARP principle.

Other changes which were introduced as a result of the POSR shutdown fault analysis were the extension and strengthening of the modes 5 and 6 Tech. Specs., and the provision of a system, independent of normal power supplies to reinstate the equipment hatch. This latter arose from the definition of various states where it was necessary to be able to restore the (open) containment to modified containment integrity within 2 hours.

4.2 Provision of Procedures and Guidance

As has already been noted, for the MCR staff severe accident guidance is included within the existing suite of operating instructions as SOI 8.8. This differs from the other SOIs in that they are not totally self contained. It was recognised that advice could be required from the TSC under some circumstances. For instance, the use of an unborated water source, gives rise to potential criticality concerns as a result of water with low boron levels being available for recirculation if the pumps were restored and the core was still essentially intact. To avoid this, the current procedure only initiates the containment fire protection system after RPV failure, which guarantees that core relocation has already taken place. Since there was no simple means of unambiguously diagnosing RPV failure under all circumstances, it was considered to be more appropriate to give the TSC this task. Analysis has been carried out and procedures developed to help the TSC accomplish this task.

The provision of additional information and guidance to both the TSC and the CESC is being pursued. Technical bases for decision making are available to Nuclear Electric both in the form of the EPRI Technical Bases Reports and the supporting reports for the level 2 PSA (which included reviews and data reports for key areas). Nuclear Electric also has access to the Westinghouse Owners Group severe accident management guidance.

Once the CESC is operational, specialist advice will be available to supplement the on-site advice. Since Nuclear Electric provided the Architect Engineering for Sizewell 'B', in-house, this includes members of the original design organisation. Both on-site and off-site support is facilitated by the ECOS system (Engineering Computer Operating System) which provides up to date information on a wide range
of plant parameters. This computer is independent of the normal control system computers and can be used to provide the basic information for additional analysis. The system can be accessed remotely as well as on site.

5. CONCLUSIONS

Procedures and guidance for dealing with severe accidents either have been or are being implemented on all Nuclear Electric Power Stations. On the operating gas cooled reactors care has been taken to implement them in such a way that it has minimum impact on the established procedures for dealing with design basis faults. Since Sizewell 'B' is the first of its kind in the UK this problem does not arise and the severe accident procedures have been integrated into the station operating procedures from the start. The development of the procedures for both systems has been supported by extensive R&D and will continue to be refined as they are used in exercises.

ACKNOWLEDGEMENT

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REFERENCES


Events of decreasing likelihood / faults of increasing severity

Design basis faults | Beyond design basis accidents | Severe accidents (degraded core)

Existing fault procedures

Symptom Based Emergency Guidelines - SBERGs

Severe Accident Management Guidelines - SAGs

Event based procedures | Symptom based advice | Benefits/disbenefits of actions dependent on Plant State
Mandatory actions | Non-mandatory actions | Advisory guidance

Fig. 1. Relationship between existing fault procedures and accident management advice
SEVERE ACCIDENT MANAGEMENT
AT THE FORSMARK OKG AND BARSEBÄCK NPP:S

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INTRODUCTION

In Sweden all nuclear power plants are equipped with filtered containment venting systems and
back-up sources to the containment spray system. These measures reduce the environmental
consequences of a severe accident and were implemented at all sites by 1988.

Emergency Operating Procedures (EOP:s) have been developed including these new features.
In this paper severe accident management at the three BWR sites Forsmark, Oskarshamn and
Barsebäck is described. All Swedish BWR:s are designed by ABB Atom. The first two
reactors at Oskarshamn and the two reactors at Barsebäck are equipped with external
circulation pumps. The third reactor at Oskarshamn and the three reactors at Forsmark are of a
later design with internal circulation pumps.

In this paper the accident management strategies and procedures are described with emphasis
on the development of the procedures and the implementation at the plant. It is important to
recognize that the development of strategies and procedures for accident management must
continue as long as we have nuclear power plants in operation. Firstly the staff at the plant
must be trained on a regular basis to be familiar with the procedures. Secondly the procedures
have to be updated utilizing new results from the research about severe accidents or due to
upgrading in the plant. In Sweden it is a tradition that severe accident research programs are
run in cooperation between the utilities and the Swedish Nuclear Power Inspectorate. During
this year the APRI (Accident Phenomena of Risk Importance) project will be completed. One
of the aims of APRI is to achieve better knowledge about severe accident phenomena to be
used in the development of strategies for severe accident management.
SEVERE ACCIDENT MITIGATION MEASURES

Since the end of 1988 all Swedish nuclear power plants are equipped with mitigating filter systems and additional water supply for the containment spray systems. This was required from the Swedish government to admit continued operation license.

The Barsebäck plants had to install these mitigation systems latest at the 1985. The reason why the Barsebäck plants had to install mitigation systems earlier than the other Swedish plants is the vicinity to Copenhagen, the capital of Denmark.

There are two types of filter equipment installed in Swedish nuclear power plants. The FILTRA system (ref. 1) in Barsebäck consists of a 100000 m³ gravel bed from quartzite rock crushed to stones 25-35 mm size. The containments in Barsebäck 1 and 2 are separately connected to the filter via vent pipes with a dimension of 600 mm. The large size of FILTRA has the advantage of a long passive function. The system requires no operator action within the first 24 hours following an accident since the gravel bed acts passively both as a filter and a heat sink. If the pressure inside the containment rises above a certain level a rupture disc will burst and the steam and gas will flow through the vent pipe into the gravel bed. The system has the capacity to handle almost all possible severe accident sequences including ATWS due to the large filter vent capacity.

The FILTRA-MVSS system (ref. 2) which is installed in the other Swedish nuclear power plants comprises a pressure relief system, multi venturi scrubber system and a moisture separator.

The pressure relief system comprises a connection from the drywell part of the containment to the venturi scrubber and a second part in which filtered gas from the venturi scrubber unit is led to the atmosphere. The system is designed for operator independent activation through the burst of a rupture disc but also available for manual activation by the operating personnel in order to obtain safer a more stable condition in the containment during and after a severe accident.

In the venturi system the gas is passing through a fine mist of water droplets which are created by the venturi nozzles. Separation of aerosols and gaseous elemental iodine takes place both in the venturi pipes and the pool water. The radioactive gas and steam is dispersed through the venturi nozzles located near the bottom of a water tank (180 m³) and then transported upwards through the water where the radioactive particles are washed out. The moisture separator collects the entrained pool water droplets.

The requirement for radioactive release in Sweden in case of accidents exceeding design basis is that less than 0.1 % of the core inventory (except the noble gases) in an 1800 MW thermal power plant may be released.

In case of a rapid pressure build up, caused mainly by malfunction of the pressure suppression system, within the containment the FILTRA-MVSS is not capable to take care of the pressure relief. In such a situation a different system is activated through the burst of a rupture disc and the steam and gas is directly released to the atmosphere. This event will happen before any core damage has occurred and the system is automatically closed after 20 minutes. In this case the ECCS systems should work properly.
Another mitigating system in terms of atmosphere cooling and cleaning is the additional water supply system to the containment. The system consists of two physically separated pipes with a diameter of 200 mm connected to the drywell sprinkling system. Water to the system is supplied from the ordinary fire water system by a diesel pump or through connection from mobile water supply.

ACCIDENT MANAGEMENT STRATEGIES AND PROCEDURES

The primary goal of the accident management strategy is to restore cooling of the damaged core and to prevent reactor vessel failure. If the disturbance develops into a severe accident and the vessel melt-through occurs the first priority is to maintain the integrity of the containment.

If an incident develops into a severe accident water is supplied as soon as possible by using the containment spray system. In order to increase the reliability of this system back-up sources of water are available and connections are prepared to the ordinary spray system. Also in case of a total station blackout the containment can be sprayed with water. Supply of water via the spray system has the following advantages: the pressure build-up by steam generation is reduced; airborne activity in the containment atmosphere is washed out and dissolved in the sump and the containment is filled with water up to a desired level to cool the core debris. The accident management strategy in Sweden is described in (ref. 3).

Water filling of the containment

In a separate project the long term effects of a severe accident have been studied for the Forsmark plant. This project is described in (ref. 4). The aims of the project were to achieve an increased knowledge about long term effects of severe accidents and to implement the results in the accident management instructions. One main result is the importance to consider long term issues also in the short term management after a severe accident. An example of a conclusion from the project which has been implemented in the accident management is the choice of water level in the containment after vessel failure. A water level slightly above the bottom of the reactor vessel is recommended.

The aim of the accident management after vessel failure is to reach a stable state of the damaged core and to minimize activity releases from the containment. In this context it is also necessary to take long term effects into consideration early in the accident. In the choice of water level the following aspects are the most important:

- The water level should be sufficiently high to cover the bottom of the reactor vessel. In this way the contribution from remaining fuel in the reactor will be strongly reduced in case of activation of the filtered venting system.

- The water level should not be higher than necessary in order to keep the water leakage from the containment as low as possible especially in the long run.

At Oskarshamn the water sprinkling in the containment from an outer source shall continue until the water level reaches the top of original core level (ref. 5). This provides cooling of remaining core material in the reactor vessel and the severe accident is brought to a safe condition. The choice of water level in the containment where to stop the sprinkling is a result from the RAMA project.
If there is a hole in the bottom of the reactor pressure vessel, but the rest of the primary system is intact the water filling of the containment can lead to that the vessel remains empty of water because of remaining gas inside the vessel. Water filling to the top of original core level provides sufficient cooling of the vessel from outside to prevent vessel failure.

Emergency Operating Procedures

At Forsmark two EOP:s (Emergency Operation Procedures) are used in parallel in the control room. The EOP for the shift supervisor is a function based flowchart procedure. The EOP for the reactor operator is a symptom based step by step procedure. The use of two EOP:s implies that the reactor operator and the shift supervisor work in a different way. This decreases the possibility that both make the same mistakes. The EOP used by the shift supervisor is based on the four critical safety functions: reactivity, core cooling, heat sink and radioactive barriers. The EOP:s are updated when a need is identified. If the plant is upgraded (for instance new instrumentation) the EOP:s are rewritten. It is important that the staff in the control room is familiar with the EOP:s. Therefore training is arranged every year with the shift crew at a full-scale simulator.

At the Barsebäck and Oskarshamn NPP:s there is only one set of Emergency Operating Procedures. The EOP:s are intended for use by the shift supervisor. The set of EOP:s consists of several flowchart procedures which are both symptom- and function-based. The EOP:s are divided into sections concerning the following parameters; reactivity, water level in the reactor, pressure in the reactor, pressure in the containment, temperature of the condensation pool and water level of the condensation pool. The EOP:s are updated on regular basis.

In a late stage of a severe accident (after reactor vessel failure) the EOP:s are of limited use for the crew in the control room. Instead the measures are taken by guidance from the emergency control center. In Forsmark an important document in such a case is a knowledge based handbook to be used by the accident management center as a basis for decisions. The handbook is not a procedure but contains general recommendations about relevant issues in case of a severe accident. The handbook is subdivided in a number of chapters where also references to further reports are given. Examples of areas where the handbook gives recommendations are:

- Use of the mitigating systems for severe accidents
- Alternatives to restore the AC power
- Estimation of the degree of core damage
- Integrity of the containment

A first version of the handbook was implemented in the organization a couple of years ago. A work has started to upgrade the handbook. Some chapters will be rewritten and in parallel new knowledge will be included. Accident management strategies and procedures at the Forsmark nuclear power plants are described in (ref. 6).

At the Barsebäck NPP a separate safety panel has been installed. The panel shows the status of the following parameters; reactivity, control rod indication, water level in the reactor, pressure in the reactor, pressure in the containment, temperature in the containment, temperature of the condensation pool, water level of the condensation pool, feed water flow (including the emergency feedwater and the emergency core cooling system) and status of FILTRA. The safety panel is not a fully separate and redundant system to the ordinary signal system but it concentrates the information from different panels in the control room into one.
Training

The operating crew are doing simulator training five to seven days every year. One part (one or two days) of this training is to study the EOP:s and train how to use it. All the plant specific simulators can deal with the EOP:s scenarios but not further in to the severe accident.

The operating crew finds it relevant and important to train how to act in a severe accident scenario. One problem is that the simulators are not capable to simulate the whole scenario after core degradation has occurred. There is a need for better training possibilities.

MEASUREMENT OF THE WATER LEVEL IN THE REACTOR VESSEL

Cooling of the core is mainly based on measuring the water level in the shroud by using conventional dp-transmitters. Findings from the TMI-accident discovered that this method could be disturbed in a severe accident situation. Using different methods of measurement is a way out of this problem.

A decision is made within the Sydkraft Company to install a core cooling monitor in the two Barsebäck plants and in the three plants at OKG. The system is called BCCM (Becker's Core Cooling Monitor).

The system (ref. 7) includes four sondpipes in which four detectors are placed at different elevations. The sonds are placed in spare positions for power range monitor sonds. The two detectors in the uppermost positions are normally used for core cooling measurement, but are automatically switched to temperature measurement mode when there is a risk for overheating the detector. The detectors placed at lower elevations, in the middle of the core and at the bottom of the reactor pressure vessel are only used for temperature measurement.

The two detectors placed in the highest elevations are used to get an early warning of decreased core cooling. The detector placed in level 2 is used as reserve for the detector in level 1. The detectors shall indicate the cooling ability of the surrounding media. This is done by heating the detector with constant electric power. If the water level decreases in the bypass channel the cooling of the detector will be reduced as the cooling changes from water to steam cooling. This will decrease the heat transfer between the detector and the cooling media approximately 300 times and the detector temperature will increase. The temperature in the detector will increase until the temperature difference between the detector and the surrounding media is sufficient to carry away the generated effect from the detector.

The detector placed in the middle of the core is used to get indication of risk for core degradation and the detector placed near the bottom of the reactor pressure vessel shall indicate if there is a risk of loosing the vessel integrity.

The measurement range for the detectors which shall indicate when there is a risk of loosing the integrity of the vessel is 0-1260 °C. The other detectors has a measurement range of 0-1000 °C.

The signals received from BCCM will only be used as information to the control room in a first phase which means that there will be no automatic functions coupled to the system. A future development could be to have automatic signals from the BCCM for depressurization of the reactor vessel and start/stop of the ECCS-pumps.
Improvement of the measurement of the water level in the reactor vessel in the Forsmark plants by implementation of a diversified system has been investigated. For Forsmark 1 and 2 the following activities have been completed:

- Specification of the requirements of an improved and diversified system
- Definition and analysis of possible accident sequences
- Identification of available methods suitable as a complement to the existing equipment
- An evaluation of the performance of different systems under normal operation and accident scenarios

Work continues to further analyse the following three diversifying methods for measurement of the reactor level in Forsmark 1 and 2:

- Direct measurement of the differential pressure in the reactor vessel.
- Collapsed water level measurement in a standpipe connected to the reactor vessel
- The EMUS (ElectroMagnetic Ultra Sound) method developed by Siemens/KWU

The first of these methods seems to be the most promising. Therefore work is going on to study this further before a decision can be taken.

CRITICAL DECISIONS IN SEVERE ACCIDENT MANAGEMENT

In an accident situation the first priority is to make the reactor subcritical and to restore the core cooling. However, if the incident develops into a severe accident and the reactor vessel fails the maintaining of containment integrity takes over as the first priority. This shift in priorities is reflected in the EOP:s. If the critical safety functions (reactivity, core cooling, heat sink and activity barriers) cannot be restored within the limits specified in the EOP:s measures are taken to mitigate the consequences of the accident. In this case maintaining of the containment integrity has the highest priority in order to keep the radioactive releases to the environment as low as possible.

Diagnosis of reactor pressure vessel failure

A crucial point in a severe accident sequence is to judge if the reactor vessel has failed or not. A question of great importance (related to vessel failure) for accident management is the extent of core damage. A method for core damage assessment has been developed for Swedish BWR:s (and PWR:s) and is described in (ref. 8). This method is based on readings from containment radiation monitors (CRM) and results from the post accident sampling system (PASS). CRM measures the total dose from radionuclides in the containment. The CRM readings are compared with precalculated values and the fractional release of noble gases is determined. The probability that the reactor vessel has failed is low if the fractional releases of noble gases are below a certain limit. In the opposite case it is generally not easy to draw any conclusions about the status of the reactor vessel. A LOCA followed by fuel meltdown and the core cooled in the lower plenum gives large releases of activity to the containment. Based on information available within a couple of hours after the initiating event this case might be impossible to distinguish from a sequence with vessel melt-through. Thus, it will be difficult to decide if the first priority is to prevent vessel failure or if it is to maintain containment integrity.
Later on, when liquid samples from PASS have been analyzed it should be possible to know if the reactor vessel has failed.

**Manual pressure release from the containment**

In the late phase of a severe accident scenario there might be necessary to decrease the pressure inside the containment through the filter. This will lead to that the pressure inside the containment will be reduced and that a large amount of the radioactivity that remains in the steam/gas phase can be removed by the filter. This action will minimise the activity release within the reactor building and thereby make it possible to do service and measurements inside the building.

The release of activity to the filter is a decision based on meteorological circumstances and also on the pressure level inside the containment.

**Saving of the ECCS**

One critical decision in case of a severe accident in the Barsebäck NPP is the question of cooling the core when the condensation pool is not cooled by the ordinary cooling system and the emergency core cooling system (ECCS) is in operation. The ECCS pumps will be in danger of cavitation due to low NPSH margin. In order to avoid cavitation the procedure is to raise the water level in the condensation pool with water from the fuel pools above the reactor. Normally the water level is 6.25 m and in case of water from the fuel pools the water level will be 12 m. If this is not enough (the cooling cannot be restored) the FILTRA will be manually activated to depressurize the containment. In order to save the pumps they have to be shut down. If the pumps cannot be reactivated due to cavitation the strategy will be to maintain the integrity of the containment.
EMERGENCY ORGANIZATIONS

The organization described below represents the Forsmark site. The organizations at the OKG and the Barsebäck site are similar with minor differences.

The emergency organization have to fulfill two demands during an accident:

- Firstly from the affected unit to handle the situation
- Secondly from the authorities and the public regarding information about the accident

In this paper we focus on the first of these two demands. In order to give an effective response at the affected unit the emergency organization should deviate as little as possible from the normal organization. An overview of the emergency organization is shown in figure below.
manager and operation manager. The plant operation manager in the ECC works in close cooperation with the unit manager. The plant operation manager is concentrated on the long-term strategy and site-related actions and the unit manager is responsible for the short-term measures at the affected unit. The time needed to update the unit manager in the beginning of the accident is minimized by using prepared check-lists in the handbook for severe accident management. This handbook is also used by the plant operation manager and his staff in the ECC.

As described earlier the EOP:s used by the shift supervisor are based on the four critical safety functions: reactivity, core cooling, heat sink and activity barriers. If the accident develops in such a way that any of these cannot be restored to normal values by the staff in the control room further actions are taken after discussion with the emergency control center. In case of reactor vessel melt-through strategies for filtered venting of the containment and supply of water to the containment are worked out by the emergency control center together with the unit manager.

CONCLUSIONS

The tradition in Sweden with a cooperation between the utilities and the Swedish Nuclear Inspectorate has shown to be very fruitful. A common strategy has been established in the area of severe accident management. With limited resources, the cooperation has made it possible to finance and obtain results from different research programs.

The general conclusion from the experience of questions regarding severe accident management at the Forsmark, OKG and Barsebäck NPP is that simulator practice of the EOP:s is necessary. Understanding and implementation of new results from the research in the severe accident area is also important. Modifications of the plants, the outcome of ongoing probabilistic safety analyses and feedback from the simulator exercises give the input to a regular update of the EOP:s.
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Severe Accident Mitigation Strategy for Operating
and Advanced VVER-1000 Reactors

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1. Introduction

The last revision of the basic regulatory document (OPB-88) requires that NPP safety assessment should deal with the so-called beyond design basis accidents (beyond-DBAs). According to definition the beyond-DBAs incorporate the initiating events concurrent with the failures and operator's errors beyond the single failure and single error principles or the initiating events not considered for DBAs.

Safety assessment during beyond-DBAs shall finally aimed at proof of non-exceeding the maximum permissible radioactivity release and restriction of exposure to the site personnel and the public to the allowable limits. The allowable limits are established by the special norms and rules.

Beyond-DBA safety shall be provided by accident management with the help of engineered features and organizational procedures. Herewith, in accordance with defence-in-depth principle the measures shall be provided both to prevent the accident progression and to mitigate the accident consequences. The measures are recognized to be effective in case when probability of the core severe damage does not exceed 10^-5 l/reactor*year, whereas probability of the maximum permissible radioactivity release does not exceed 10^-7 l/reactor*year.

In accordance with the regulatory document requirements for safety substantiation during beyond-DBAs it is necessary to consider all the phenomena and processes occurred during the accidents and having the governing effects onto the integrity or damage rates of the barriers for radioactive products retention and onto the reliability of
engineered features used for accident management. For future reactors being designed, based on the accident analysis it is possible to reveal design bases for the engineered features used for accident management and the relevant management procedures. For existing VVERs whose designs are in conformity with the early revisions of the regulatory documents such analyses result in the necessity of some modifications of NPP.

2. General strategy of VVER accident management

Beyond-DBA accident measures are also proceeding from the defence-in-depth principle and from the engineered features available at NPP. In this case the defence-in-depth principle is expanded beyond the design basis for NPP into the domain of beyond-DBA conditions including the spectrum of severe fuel damages.

Accident management is oriented, mainly, for the operative personnel's actions. Aim of actions is to stop accident progress or to restrict radioactive releases. In the course of accident management the operative personnel shall try

1. To prevent the core damage.
2. To stop the process of core damage if it has started and to retain the core inside the reactor vessel.
3. To retain the containment integrity as long as possible.
4. To minimize the radioactivity releases.

To achieve the objectives one should use all the engineering features being available for the operative personnel, i.e. normal operation systems, safety systems and operator support systems. Success can be achieved owing to conservative design and layout provisions provided the most important safety functions (herein after referred to as the critical ones) would be restored. The following safety functions are singled out as the critical ones:

**Subcriticality**

Function content: (1) to prevent the unallowable reactivity variations; (2) to trip the reactor, if necessary; (3) to maintain the reactor under safe shutdown condition, i.e. in the state of sufficient subcriticality during and after the condition required the reactor to be tripped.

**Core cooling**

Function content: (1) to remove the residual heat during normal operation, operational occurrences and accidents not related to loss of primary coolant pressure
boundary integrity; (2) to remove the core residual heat after loss of primary coolant pressure boundary integrity in order to mitigate the fuel rod damages; (3) to remove the core residual heat during the beyond-DBAs in order to stop the process of core damage.

*Primary coolant pressure boundary integrity*
Function content: (1) to prevent the primary component structures from the unallowable thermal and mechanical loads; (2) to prevent or to mitigate the reactor pressure vessel failure during the core melt progression.

*Coolant inventory in the primary circuit*
Function content: (1) to maintain coolant inventory in the primary circuit that is needed to provide circulation including natural circulation, under the conditions with primary coolant pressure boundary intact; (2) to maintain coolant inventory in the primary circuit that is adequate for core cooling in case of design leaks from the primary circuits.

*Ultimate heat sink*
Function content - to provide heat removal from the reactor plant to the ultimate heat sink.

*Containment integrity*
Function content - to prevent the containment from the unallowable thermal and mechanical loads and to avoid the uncontrolled radioactivity releases.

*Restriction of radioactivity releases*
Function content - to restrict the radioactivity releases to the allowable limits.

According to OPB-88 requirements the on-line display systems shall be used to provide the operative personnel with the generalized information on safety status of the reactor plant and NPP as a whole. Thus, a set of measured and calculated characteristics shall be provided to reflect the unit safety status. In the operator’s support systems these parameters are defined as the safety parameters.

Safety parameters include the characteristics indicating status of the critical safety functions. Operative personnel shall undertake the actions on restoration of the critical safety functions having information on safety parameters. Being within the allowable limits, the relevant safety parameters also characterize integrity of the corresponding safety barrier. In particular, the safety parameters include the measured parameters that
directly testify to the fact of failure of the corresponding safety barrier. The safety parameters to be used by the operative personnel are as follows:
- reactor power,
- reactor reactivity,
- margin to fuel melting,
- maximum burnup,
- DNBR,
- primary coolant radioactivity,
- primary pressure,
- containment radiation level,
- secondary coolant radioactivity,
- pressurizer coolant level,
- secondary pressure,
- SG level,
- containment pressure,
- containment temperature,
- hydrogen concentration in the containment rooms,
- radiation level outside the containment.

The most of the above safety parameters are measured by the standard monitoring and control systems of the unit. Nowadays, the generalized parameters are defined in the restricted scope and are not well systematized. Under these conditions the NPP safe operation is achieved by combination of activities performed by the automated control systems and functions performed by the operator.

3. Accident management for existing VVER-1000

Chernobyl disaster, new national safety regulations and NPP operation experience have facilitated to the development of the balanced accident management for existing VVER-type reactors. A number of first-priority measures have been developed to provide the better understanding of severe accident role and phenomenology and to implement at existing VVERs some technical means for severe accident management. Main of these measures are as follows:

- Probabilistic safety analyses in order to define the main severe core damage contributors and to propose the accident management measures.
- Beyond-DBA analysis in order to understand severe accident sequences and assess efficiency of the accident management measures.
- Equipping of the operating units with the standard reactivity monitoring systems.
- Development and implementation of the additional means for SG makeup from the reliable sources in order to enlarge time duration of the SG emergency makeup above 8-10 hours.
- Implementation of the containment hydrogen monitoring and removal system.
- Development and implementation of the diagnostic system to monitor the status of the safety important systems.
- Modification of control and monitoring system in order to provide the automatic and/or computer-aided diagnostics of the engineered feature status.
- Elaboration and implementation of the automated radiation monitoring system.
- Development and implementation at NPP of the information support system for personnel.
- Development and implementation of the emergency centers for operative technical assistance to the plant personnel in case of accident.

Alongside with the above and other measures that require a certain time to be implemented, accident management guidelines were developed and implemented at nuclear power plants. Now there are two documents at NPP which should be applied by personnel depending upon the NPP status: "Instruction on accident liquidation" and "Instruction on beyond design basis accident management". The first document deals with design basis accidents according to the list developed in the frame of NPP design and agreed upon with the regulatory body. In this case the objective of the personnel actions is to return plant parameters and equipment state into the limits of the safe operation.

The second document includes the recommendations developed currently for beyond-DBA management on the basis of PSA level 1 results; these recommendations are the first stage in elaborating the symptom-oriented emergency manuals. The document describes also the organization structure of executive management on beyond design basis accident (the main tasks of the special staff, the rights, duties and responsibilities of its members and of the plant leadership) and gives some non-design ways of equipment use for the safety functions fulfillment. When elaborating the recommendations the following accidents are reviewed that make the essential contribution to the frequency of the core severe damage and radioactivity release:
- small break LOCA with high and low pressure ECCS failure;
- medium break LOCA with high and low pressure ECCS failure;
- large break LOCA with high and low pressure ECCS failure;
- primary-to-secondary leak with subsequent non-closure of the SG relief/safety valve;
- rupture of the scheduled cooldown line without closing of the localization valves in the containment penetration;
- large leak from the primary circuit without closing of the ventilation system valves;
- long-term NPP blackout with failure of safety system diesel-generators to start;
- steamline rupture in the SG non-isolable part with failure of long-term removal heat system from the reactor plant through the scheduled cooldown line and normal heat removal system through the SGs;
- rupture of the main steam header with failure of heat removal system from the reactor plant (low pressure ECCS) and emergency feedwater pump system;
- deterioration of the secondary heat removal with failure of the reactor plant heat removal system.

Let's consider the main arrangements of these recommendations for typical beyond-DBAs as they are given in the instruction on beyond design basis accident management.

**Small break LOCA with failure of ECCS active part**

A peculiarity of small break LOCAs is the significant influence of the SG heat sink upon the primary circuit processes and upon the core cooling. Small break LOCAs may appear in case of main circulation circuit unsealing due to ruptures of the associated system pipelines or during the pressurizer valve opening.

Small break LOCAs concurrent with the ECCS active part failure result in non-fulfilment of the critical function related to maintenance of the primary inventory being sufficient for core cooling. Core uncovery occurs and core cooling fails that leads to exceeding the maximum design limit for fuel rod damage and core melting. To avoid this the operator shall provide water supply into the reactor and his actions shall be directed to reveal operability of the systems enabling to supply water for core cooling, to actuate these systems and to restore operability of the ECCS active part. In addition it is necessary to decrease loss of coolant into break due to reactor plant cooldown at a maximum permissible rate through the secondary circuit.

The main signs of the accident are as follows:
- connection of all groups of the pressurizer electrical heaters;
- closing of the primary blowdown controller valves and total opening of the makeup controller valve;
- pressure rise in the containment.

The additional signs of this accident are as follows:
- actuation of the reactor emergency protection in response to the signal of primary pressure decrease;
- formation of the signal "difference between the primary saturation temperature and the maximum temperature in any hot leg is $T < 10^\circ C$";
- safety feature actuation in response to the signal of $T < 10^\circ C$;
- closing of the containment localization valves in response to the signal of $T < 10^\circ C$;
- connection of the spray system when the containment pressure rises to 0.13 MPa.

Calculation analysis of small leak with $D_{nom} = 50$ mm in the cold leg has shown that in about 1.5 h since the accident initiation the core starts uncovering if the operative personnel failed to provide water supply into the primary circuit at this stage. The primary pressure stabilizes above the ECCS hydroaccumulator operation pressure and is defined by the secondary pressure. Depending upon leak size the ways of primary pressure decrease in order to put the ECCS hydroaccumulators into operation are recommended as follows:
- usage of the emergency gas-removal system by opening the corresponding valves;
- pressurizer water injection from the makeup system;
- reactor plant cooldown through the SG including steam dumping into the atmosphere.

Putting the hydroaccumulators into operation enables to use the additional time of about 3 h for core cooling to be restored. At estimates, in about 4 h since the moment of putting the hydroaccumulators into operation the core starts melting. Actions on boron solution supply into the core shall also be continued at the subsequent accident stage.

**NPP blackout**

NPP de-energization results in loss of alternative current power supply to the station auxiliaries from the normal operation on-site sources (turbogenerator) and off-site sources (power grid). De-energization leads to loss of normal heat removal from the primary and secondary circuits caused by disconnection of the reactor coolant pumps, turbine trip and impossibility of heat removal from the reactor plant through the turbine condenser.

Accident with long-term NPP de-energization and failure of three diesel-generators to start results in non-fulfilment of the critical functions related to the core residual heat removal and ensuring the primary coolant inventory and, consequently, to core uncovery and fuel damage.

Operative personnel's actions on preventing the accident progression and mitigating the accident consequences shall result in restoration of power supply at least in one safety system channel. The primary signs of this accident are:
- sudden secondary pressure rise with possible BRU-A operation;
- RCP dis-connection, reactor emergency protection actuation, TG stop valves closing;
- failure of three diesel-generators to start.

The processes typical of the given accident can be considered as the additional signs. By the results of estimates, in about 7-10 min since the accident initiation the pressurizer safety valves start to operate and the uncompensated coolant discharge starts, in about 2 h the SG emptying begins. The primary processes run at a high pressure that corresponds to the pressurizer safety valve operation, in this connection the measures are recommended to put into operation the ECCS hydroaccumulators first of all. For this purpose it is recommended to decrease the primary pressure using the emergency gas removal system and pressurizer safety valve.

When the operator's actions on restoring operability of the active safety systems are not effective to avoid the reactor vessel failure, it is required that the primary pressure should be decreased by the means available.

4. Mitigation of severe accidents in advanced VVER designs

TMI and Chernobyl accidents have resulted in more stringent safety requirements for new NPPs stated in Russian safety guides. Many different advanced reactor concepts were proposed in Russia with the main purpose to decrease the probabilities of severe core damage and excessive release by factor 10...100. At present, there are a few advanced VVERs under development. VVER-640 (V-407 design), NPP-92 (V-392 design) and NPP-91 (V-428 design) developed by EDO Gidropress are now considered as the most ready to construction. It is planned to erect the first units with these reactors by the year 2000-2002. These designs reflect two trends in the advanced VVER development.

The first trend is gradual, evolutionary improvement of standard VVER-1000 (V-320 design). This trend is represented by the Russian-Finnish VVER-91 concept. The changes of standard V-320 NSSS include: advanced VVER-1000 reactor itself; improved SG; four-train safety systems; double containment; filtered containment depressurization system; special system to cope with primary-to-secondary leak; implementation of systematic severe accident management, etc. More details on this concept and its severe accident management are given in separate paper by Mr. Antropov G.A.

The second trend makes use of relatively new technical decisions to meet new safety requirement being foreseen by the year 2005. This trend is represented by medium size V-407 and by large size V-392 reactors. Principles of the inherent safety are more widely used in these designs. Reasonable combination of passive and active safety systems is
used to assure higher indices of NPP safety. Some important characteristics of these reactors are as follows.

<table>
<thead>
<tr>
<th>Parameter</th>
<th>V-407</th>
<th>V-392</th>
</tr>
</thead>
<tbody>
<tr>
<td>Thermal power, MW</td>
<td>1800</td>
<td>3000</td>
</tr>
<tr>
<td>Electrical power, MW</td>
<td>640</td>
<td>1000</td>
</tr>
<tr>
<td>Number of loops</td>
<td>4</td>
<td>4</td>
</tr>
<tr>
<td>RCS pressure, MPa</td>
<td>15.7</td>
<td>15.7</td>
</tr>
<tr>
<td>Reactor inlet/outlet temperature, C</td>
<td>297/327</td>
<td>290/320</td>
</tr>
<tr>
<td>SG pressure, MPa</td>
<td>7.06</td>
<td>6.3</td>
</tr>
<tr>
<td>Average linear power in fuel, W/cm</td>
<td>106</td>
<td>166</td>
</tr>
<tr>
<td>Specific core power, kW/l</td>
<td>65</td>
<td>107</td>
</tr>
<tr>
<td>Number of control rods</td>
<td>121</td>
<td>121</td>
</tr>
</tbody>
</table>

These reactors are provided with the active and passive features to protect safety barriers and/or mitigate the consequences of their damage. Safety systems of VVER-640 include:

- **Reactor protection system.** This system contains 121 control rods and their total worth is sufficient to scram reactor and to keep it in subcritical condition even if all the boron will be removed from coolant and coolant will be cooled down up to 20 C.

- **Passive emergency core cooling system.** This system consists of hydroaccumulators and low pressure or opened to containment tanks with borated water (located up to 30 m above the core). After the RCS, hydroaccumulators and low pressure tanks become empty, the water level in the special pool around the RCS is higher than reactor inlet/outlet nozzles. The opening of depressurisation valves leads to natural circulation via the reactor downcomer, inlet plenum, core, upper plenum and the pool around RCS. The heat removal from the pool is assured by evaporation of its water and condensation of steam on the steel containment wall and other structures.

- **Passive heat removal system from SG secondary side.** The steam from SGs is directed to surface heat exchanger submerged into special heat-accumulating tanks located on the concrete containment shell. The condensate is returned to SGs.

- **Depressurisation valves on primary loops (SDV).** They are closed normally by pressure of the coolant. If primary pressure decreases up to 0.3-0.5 MPa, they are open passively and remain in open position further on.
- Containment heat removal system consisting of two cooling bands on the steel shell outer surface.

The advanced VVER-1000/V-392 reactor has a number of systems similar to the above described. In particular, reactor scram system contains 121 control rods to assure subcriticality up to 100 C without boron solution injection. Passive part of the emergency core cooling system consists of high pressure (6.0 Mpa), low pressure (1.2 MPa) hydroaccumulators and open to containment pools (refuelling and reactor internals inspection pools). Passive heat removal system is similar to V-407 but uses air heat exchanger installed outside of containment The system is capable to remove all the residual heat during very long period. Quick boron supply system is provided as a back up mean for reactor shutdown. This system consists of 4 tanks with boron solution installed on the bypasses of main coolant pumps. Water inventory and boron concentration in the system assure the same worth as for the reactor scram as by control rods.

All the above systems ensured that severe core damage probability for both designs is less 10-6 per reactor*year. Nevertheless some engineered features are provided to mitigate the relevant consequences. The special system with filter is provided to avoid containment overpressurization and decrease radioactivity release from the containment into environment. Hydrogen catalitic recombiners are installed in the reactor building compartments to avoid the formation of flammable gaseous mixtures. It is supposed to keep the core melt within the reactor pressure vessel or in the concrete reactor cavity or in the advanced compartment. A few concepts of "core catcher" are now being studied for this purpose.
CONCEPT OF SEVERE ACCIDENT MANAGEMENT FOR NPP-91 DESIGN WITH VVER-1000 REACTOR

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OECD SPECIALIST MEETING ON
SEVERE ACCIDENT MANAGEMENT IMPLEMENTATION

JUNE 12–14, 1995
CONCEPT OF SEVERE ACCIDENT MANAGEMENT FOR NPP-91 DESIGN WITH VVER-1000 REACTOR

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INTRODUCTION

The NPP-91 concept at first have been worked out for the conditions of Finland, and it participated in the tender for construction of the next power unit at the NPP in Finland. Therefore, from the very beginning of the concept development in addition to the Russian safety norms in force the design requirements of Finland and IAEA recommendations were taken into account. For proper interpretation of the Western norms and some support in the design "Atomenergoexport"(Moscow) enlisted IVO International from Finland as a consultant.

The regulatory authorities of Finland (STUK) made a preliminary review of the submitted design documentation and found no fundamental reasons to prevent the NPP-91 licensing in Finland.

The NPP-91 design concept is consistent with the safety objectives for evolutionary type advanced reactors. Maintaining the proven features of operating VVER-1000/320 plants, avoiding needs for new technology development and adopting the proven design solutions from other NPPs will provide a sufficient basis for NPP-91 with VVER-1000/428.

Both preventive and mitigative accident management features of NPP-91 with VVER-1000/428 have been strengthened. In particular, there is a strong containment to assist in coping with potential severe accidents. This is consistent with the trend in the international practice for evolutionary type advanced reactors.

1 BRIEF DESCRIPTION OF NPP-91 CONCEPT, MAIN DESIGN FEATURES AND SAFETY DESIGN PRINCIPLES

The NPP layout provides for a monounit arrangement, the power of every monounit is about 1000 MW(e). The design lifetime of the NPP main equipment is 40 years. The power unit includes a reactor plant with a VVER-reactor and a turbine plant. The power reactor VVER-1000 (reactor plant V428) is a vessel-type PWR reactor. The turbine plant includes a steam turbine and generator, mounted on a single foundation with the turbine.
Main parameters of the power unit

Reactor unit

- Thermal power, MW(th): 3000
- Primary circuit pressure, MPa: 15.7
- Coolant mean temperature, °C:
  - reactor outlet: 321.7
  - reactor inlet: 292
- Uranium quantity, t: 75
- Average burnup fraction, MWD/kg uranium: 43
- Average fuel enrichment, % weight: 2.57
- Average fuel enrichment at makeup by uranium-235 isotope, % weight: 4.0
- Number of loops: 4
- Steam generator type: horizontal

Turbine plant

- Power at a cooling water temperature of 11 °C, MWe: 1070
- Rotation speed, rpm: 3000
- Number of cylinders, HP+LP: 1+4
- Steam flow rate, kg/s: 1630.2
- Rated steam conditions:
  - pressure, MPa: 6.08
  - temperature, °C: 276.4
  - dryness degree: 0.995

The defence-in-depth principle is well taken into account in the NPP-91 design with VVER-1000/428 reactor. All levels of protection, except for the last one for emergency measures introduced by local authorities, have been strengthened by increasing the design margins, inherent plant features, safety and protection system capabilities and containment protection means. Therefore the reliability of the main safety functions: shutdown capability, cooling the fuel and confining the radioactive material has been remarkably improved. These safety functions serve as a basis for avoiding accidents during operation and controlling accidents if they occur.

The design uses the deterministic approach supposing that the design ensures the safety under normal operation and any initiating event considered in the design with allowance for the single failure principle. Besides, the design incorporates measures concerning the management of severe accident and mitigation of their consequences.

The plant will have the following buildings important for nuclear safety: reactor building, steam cell, safety building, control building and diesel building.
Reactor building

A double containment concept is adopted. The intermediate space between the containments is equipped with a vacuum ventilation system with the filtered exhaust. The primary containment will be cylindrical, dry, full-pressure type, equipped with a spherical dome. The primary containment will be made of a pre-stressed concrete. The base slab will be made of a reinforced concrete. The primary containment will be dimensioned against a classical loss-of-coolant accident. The secondary containment will be made of an ordinary reinforced concrete. The secondary containment provides a physical protection for the primary containment.

All spent fuel will be stored in the refuelling pool of the reactor building. The storing capacity of the refuelling pool is designed for about ten reloads and a place for the reactor core evacuation is reserved.

Steam cell

The building places the rooms for the overpressure protection system of the steam generators, emergency feedwater system and demineralized water supply system. The residual heat of the reactor will be removed with the help of the systems in the steam cell at the transients, where the primary circuit remains intact due to dumping out the steam to the atmosphere through controlled safety valves and by supplying water from the demineralized water tanks into the steam generators with the help of the emergency feedwater pumps.

Safety building

The purpose of the safety building is to provide for a protection for:
- active part of the emergency core cooling system (ECCS)
- borated water storage tanks
- component cooling system and service water system
- containment spray system
- emergency boration pumps
- vacuum ventilation system for the annular space.
- containment filtered venting system.

The systems will have a back-up electric power supply system. The power supply for the safety building is provided from the switchgear plant of the diesel building.

Control building

The building places the rooms for the instrumentation and control system equipment with associated power supply systems as well as the power supply systems for the auxiliary and reactor buildings.

The purpose of the building's spaces are as follows:
- to provide all rooms for all instrumentation and control equipment of the plant
- to provide rooms for the main control room, emergency control room and rooms for the computer systems
- to provide office spaces for the operating personnel of the control room
- to provide rooms for the power supply system of the primary circuit and its auxiliary systems
- to provide the DC power supply system for the plant I&C system.

**Diesel building**

The equipment of the emergency electric power supply system with switchgear plants are located in this building.

**Safety system structure**

The active safety systems consist of 4 completely independent trains. The train capacity, quick action and other characteristics were selected proceeding from the condition to ensure nuclear and radiation safety under any initiating events considered in the design. A high degree of the trains physical separation is achieved thanks to the location of the safety system trains in separate rooms and separate pits within the containment.

The safety system trains are isolated from each other by fire-resistant physical barriers along their whole length, including interconnections between the buildings.

The following systems important to safety are provided in the NPP–91 design:

**Protection systems**
1) Reactor preventive and emergency protection system
2) Emergency core cooling system, ECCS
3) Emergency boration system
4) Residual heat removal system
5) Emergency feed water system
6) Overpressure protection system
7) Emergency gas removal system
8) Primary–to–secondary circuit leak discharge system
9) Borated water storage tank system
10) Emergency water discharge system

**Localizing systems**
1) Double–shell containment
2) Containment spray system
3) Containment hydrogen control system
4) Containment filtered venting system
5) Containment isolation system

**Supporting systems**
1) Emergency power supply system
2) Nuclear component intermediate cooling water system
3) Secured service water system
4) Ventilation systems of the safety system compartments
5) Other systems

Control systems
1) Engineered safety features actuation system, ESFAS
2) Reactor scram system

2 SEVERE ACCIDENTS ASSESSMENT

A strategy to assess and manage the severe accidents has been developed. The Finnish requirements concerning severe accidents have been applied as a starting point. Because of the comprehensive nature, the strategy is called the Consistent Approach to Severe Accident Assessment and Management [1]. The focus of the approach is ensuring the containment integrity with a high level of confidence. In case of a major containment failure, it is impossible to satisfy the requirements concerning large releases.

The basic approach is to ensure that the accident prevention would be afforded by all the practical means. The design goal is to obtain the core damage frequency less than $10^{-3}$/r-yr. This shall be demonstrated with Probabilistic Safety Analysis of level 1 (PSA level 1).

For some of the core damage sequences obtained from the PSA level 1, it is impossible to demonstrate the containment integrity in a reliable way, and they would necessarily lead to large releases. Such sequences are:
- containment by-pass sequences
- high-pressure core melt sequences
- reactivity initiated core damage sequences.

Consequently, the requirement is that the probability fraction of such sequences would be negligible. In this context, a limit of $10^{-3}$ is selected for their fraction.

The next step is to study all the physical phenomena during a severe accident, which would potentially challenge the containment integrity. Such phenomena can be divided into the fast loading of the containment:
- in-vessel steam explosions
- hydrogen burns
- recriticality of the core after control rod melting
- ex-vessel steam explosions
- steam generation spikes
- Direct Containment Heating (DCH)
- missiles

and into the slow loading due to:
- steam generation
- generation of noncondensable gases
- core-concrete attack
- temperature loading to the containment structure.
Again, to ensure the negligibility of large releases, the conditional probability of any phenomenon to endanger the containment should be clearly below $10^{-2}$.

Assuming that the protection of the containment against large releases has been successful, the amount of radioactive releases should be analysed for the case of the leak-tight containment. The leak-tightness can be lost by operation of the filtered venting system or by the larger than the design containment leakage rate. Here, the absence of acute radiation damages to nearby inhabitants and of long-term contamination with less than 100 TBq of Cs-137 should be demonstrated.

The integrated severe accident analysis codes (STCP, MAAP3.OB) have been applied to study a number of beyond design basis and severe accident sequences such as:
- total loss of AC power (for 24 h)
- total loss of feedwater
- large break LOCA without emergency core cooling
- small break LOCA without emergency core cooling
- large break LOCA and blockage of coolant recirculation
- primary-to-secondary leakage without isolation.

These analyses provide the time frame and general understanding of severe accident progression at the NPP-91 plant. All the physical phenomena have been considered separately to evaluate their importance, to obtain physical understanding and to estimate phenomenological uncertainties. The obtained understanding is then utilized in developing the severe accident safety functions for the plant concept.

3 THE DESIGN PROVISIONS AGAINST SEVERE REACTOR ACCIDENTS

Initial prerequisites for development of measures aimed at the management of severe accidents were based upon the following principles:

- NPP systems and components have considerable conservative design margins and keep, at least, their operability under conditions of a severe accident
- most part of possible severe accidents, according to the results of the studies and analyses, develops relatively slowly and there is enough time at disposal of the operational personnel to interfere in the progress of events in order to stop the process, recover fuel cooling and preserve the integrity of the containment.

The provisions against severe reactor accidents have been made. However, the reactor core melt accident is not a design basis of the containment, because there are no internationally accepted design criteria for those kinds of accidents:
- The design pressure of the containment exceeds about 30 % of the maximum calculated pressure at a loss-of-coolant accident.
- Below the reactor pressure vessel there is a 3 m thick reinforced concrete structure above the containment liner.
- The steam flow section from the reactor cavity will be about 6.5 m².
• The total floor area of the reactor cavity and the entrance room is about 80 m². The free height in the reactor cavity will be about 2.2 m.
• The bottom floor of the reactor building will have floor inclinations in the direction of the reactor cavity. The primary circuit water volume is capable to flood the reactor cavity.
• The water in the reactor internals inspection wells can be used as reserve cooling medium for molten reactor core – if so required. The draining of those wells can be arranged on the bottom floor of the reactor building.

4 REQUIREMENTS FOR SEVERE ACCIDENT MANAGEMENT

4.1 General approach

The up-to-date approach to the NPP safety is based on the principle of defence-in-depth. This principle defines the availability of several physical barriers (fuel matrix, fuel rod cladding, primary coolant circuit boundary and containment) along the path of propagating radioactive products. The principle also defines several protection levels providing for the protection of physical barriers against damages and protection of the population and the environment against radiation effects, if the barriers are found to be damaged.

Severe Accident Management (SAM) creates the last level of such protection and comprises the actions aimed at prevention of transition of any accident to a severe form and mitigation of severe accident consequences.
For those actions any available serviceable technical means intended at normal operation to provide for the safety assurance under design basis accidents or specially intended to mitigate severe accident consequences are used.
The SAM measures are intended for prevention of the core damage, retention of the molten core in the reactor vessel, and if the core damaged, for maintaining the containment integrity and limitation of radioactive releases into the environment.

The mitigating measures include the personnel’s actions resulting in the mitigation of consequences of events, including actions on retention of radioactive products within the provided barriers and the limitation of possible radioactive releases into the environment.
It should be noted that for proper diagnostics the operating personnel shall reliably identify the state of the plant and to choose correctly the most effective measures on management of severe accidents which are provided in the design decisions.

4.2 SAM Measures

When planning the SAM measures the following levels are considered:
- prevention of the core damage
- prevention of the reactor vessel melt-through
- prevention of the containment failure
- reduction in radioactive releases into the environment.

For every level the safety goals are specified, and there are defined the safety functions which
are required to be satisfied for reaching the assigned goals. The appropriate actions of the personnel on performing their functions required are envisaged and the criteria of successful fulfilment of those actions are formulated.

The safety goal for prevention of the core damage ensures the decay heat transfer from the core. The safety function is to remove the decay heat through the secondary circuit or to supply the core with a sufficient cooling water flow rate being no less than the flow rate of steam generated in the core. The criterion of successful actions of the personnel is non-exceeding the coolant temperature at the core outlet above the saturation temperature. In case the specified actions of the personnel are not effective, that is characterized by increase in the reactor outlet temperatures, an operator takes some additional measures on the primary pressure decrease with opening all possible sources of coolant discharge from the primary circuit.

To prevent the reactor vessel melt-through, the required safety function is to supply the cooling water into the reactor prior to the corium falls onto the vessel bottom.

To prevent the containment failure, the safety functions are as follows:
- containment pressure decrease by the spray system
- controlled venting of the containment into the environment through the discharge filter
- hydrogen control
- providing the corium coolability on the containment floor
- ensuring the subcriticality of the corium.

4.3 Equipment and instrumentation for SAM

The SAM implementation presupposes the ability of the main and auxiliary systems to perform the required functions. Therefore the SAM implementation planning starts with the defining all plant systems (including the non-safety related systems) which may be used for accident prevention or for mitigation of its consequences. This work shall include also bringing out the back-up systems capable to perform the same functions.

Another aspect of the equipment backing up under severe accident conditions is the usage of the equipment and materials which may be obtained from other parts of the site or from outside.

For the correct determination of the plant state and level of accident severity, the NPP shall be equipped with the appropriate measurements. The instrumentation trains used during the accident management shall be serviceable under the postulated conditions of severe accidents and designed for the appropriate range of parameter measurements.

For example, for control of the core melt condition inside and outside the reactor vessel gamma radiation sensors are needed. A dose rate of gamma radiation which shall be recorded by the sensors will be within a range from $10^2$ to $10^3$ Gy/h depending on the sensor position. The concrete cavity temperature-sensing elements, which give the information on the condition of the corium outside the reactor vessel, shall be designed for the same parameters.
The main SAM measurements include the instruments and communication lines for measurement and indication of:
- neutron flux
- temperature (core outlet, primary and secondary circuits, containment)
- water levels (primary and secondary circuits, containment)
- pressure (primary and secondary circuits, containment)
- activity (secondary circuit, containment)
- condition of the corium inside and outside the reactor vessel (temperature, location, criticality)
- reactor vessel temperature
- concrete cavity temperature
- composition of the containment atmosphere (for example, hydrogen concentration);
- state of the safety systems.

The instruments and communication lines shall be completely independent and capable of functioning under the condition of the plant total blackout during 24 hours. One of the methods of meeting this criterion is the delivery of portable generators to the site for recharging the storage batteries. A possibility of destruction of a part of instruments during a severe accident shall be taken into account.

4.4 Preventive measures and instructions

4.4.1 Required information

On the basis of the knowledge gained from the severe accident research, the SAM Guidelines and Procedures shall include measures to terminate or to slow down the severe accident progression at any stage, or to prevent a transition of an accident to the severe form. Those procedures are to be aimed at the maintaining or restoration of the safety functions using the safety systems as well as the normal operation plant systems.

The SAM procedures shall be developed to provide a direct link between the plant status and the personnel's actions.

The plant status is governed by the state of the primary circuit, secondary circuit and the condition of the safety systems.

The following four parameters, as minimum, may be applied to characterize the core state:

- boiling margin defined as a difference between the coolant temperature at the core outlet and the saturation temperature
- core criticality
- water level in the reactor
- coolant temperature at the core outlet.

Information on the mentioned parameters allows an operator to monitor the core state and efficiency of heat removal from the core.
To describe the state of the primary and secondary circuits the following parameters can be used, as minimum:
- the above parameters as well as the primary pressure and the pressurizer level
- water levels in steam generators
- activity in steam generators
- pressure in steam generators
- containment pressure.

Information on the above parameters allows an operator to monitor the integrity of the primary circuit and steam generators as well as the efficiency of heat removal through the secondary circuit.

The following three parameters, as a minimum, may be applied to describe the state of containment:
- activity in the containment
- pressure in the containment
- temperature in the containment.

Information on the mentioned parameters allows an operator to monitor the degree of challenging the integrity of the containment.

4.4.2 Prevention of core damage

The prevention of core damage and melting is possible by restoring the residual heat removal function. Residual heat removal may be performed via the secondary circuit or by supplying the cooling water into the primary circuit.

A time which operator has for prevention of the core damage with a possibility to use the secondary circuit is determined by the time of breakdown of the coolant circulation through the loops. For example, under the plant blackout conditions with cessation of feedwater supply into SG, full breakdown of circulation through the loops happens at 4600 s.

An operator has specified time for a possibility of carrying out heat removal via the secondary circuit. Water supply from one emergency feed water (EFW) pump in the specified time will be sufficient for prevention of the core damage. Any other secondary side "bleed&feed" possibility shall be instructed.

If impossible to use the secondary circuit for residual heat removal, an operator is to use the primary circuit "bleed&feed" procedure with discharge of the primary coolant into the containment. It is necessary to secure water supply from any pumps available (volume and boron control system, high pressure safety injection system, low pressure safety injection system). For the purpose of the primary system depressurization, the design option of a separate relief capability with the dedicated valves shall be considered.

A time which the personnel has for corrective measures for prevention of the core damage is determined by the core uncovering. For example under the conditions of the plant blackout with cessation of feedwater supply into SG, the onset of core uncovering resulting in the fuel
heatup starts at 5100 s. In case of small-break LOCA of DN 25 mm and DN 50 mm with loss of ECCS injection, the onset of core uncover happens in 7 hours and 1.5 hours, respectively. The operator actions for the reactor system cooldown via the secondary circuit result in the primary pressure decrease down to the preset values of operation of high-pressure and low-pressure ECCS pumps.

In addition to the actions on restoration of the heat removal function, the operating personnel shall be sure that the core is in the subcritical state and that the measures undertaken cannot violate this state, i.e. if water is supplied into the reactor, then one shall be sure that it has the necessary boric acid concentration. Furthermore, if measures are undertaken to cool down the reactor plant, then one shall be sure that such a boric acid concentration in the primary circuit is created and that the core subcriticality for the "cold" state is ensured.

4.4.3 Prevention of reactor vessel melt-through

If the effective core cooling cannot be restored, the following factors threaten vessel integrity:
- worsening of strength properties of the vessel material due to its heating
- lessening of bottom thickness during its melting by the corium
- dynamic loads due to steam generation during interaction of the corium with water in the reactor
- excessive stresses in the vessel during possible supply of ECCS cold water into the heated vessel.

Within the considered period of an accident those effects may exist both separately and in combination of some of them.

Excessive heating-up and variation of the core position in the reactor vessel are the main causes of physical phenomena threatening the vessel integrity. Therefore, the operation personnel continues the actions directed to restoration of serviceability of the systems which can supply water into the reactor.

Timely actions of the personnel as to supply cooling water from the ECCS active systems permit to restore effective fuel cooling and to prevent damage of the reactor vessel. Here again, it must be ensured that the core or corium remains subcritical in various stages of adding water to it.

If the actions undertaken are not effective and reactor vessel melt-through is inevitable, it is necessary to undertake measures on reactor depressurization when the core outlet gas temperature reaches 400°C.

The primary depressurization system operation shall decrease the primary pressure down to 1 MPa in order to have the reactor vessel melt-through at a low pressure.

The large break LOCA calculation assuming failure of the ECCS active part has shown that release of melt beyond the reactor boundaries takes place in 1h 50min from the LOCA initiation.

The calculation of small break LOCA (DN 25) with failure of the ECCS active part has shown that release of corium beyond reactor boundaries happens 8h 50 min from the accident initiation.
The large break LOCA calculation assuming failure of the recirculation mode has shown that release of melt beyond the reactor boundaries happens in 3h 8min from the onset of the conditions.

4.5 Prevention of containment failure

4.5.1 Prevention of early containment failure

The early possible containment loading phenomena challenging the containment integrity were listed in Section 2.

The design measures taken for exclusion of the early failure of the containment are reduced to the following:

- concrete structures of the compartments wherein the primary equipment is housed are made in such a way that to exclude direct impacts of media and materials to the containment
- the system of corridors between the compartments and rooms of the containment secure damping of possible impact loads to the containment
- availability of a great quantity of water in the containment prevents a quick heating of the structures and accumulate the core energy for the time being
- the primary system depressurization prevents the loading caused by high-pressure melt ejection from the reactor vessel
- hydrogen control system.

The hydrogen control system shall be designed to account for oxidation of metals in the reactor core. The hydrogen concentration monitoring system shall be required.

4.5.2 Prevention of late containment failure

The causes of the late pressure increase in the containment during severe accidents are as follows:

- steam generation as a result of a long-term water evaporation from the containment floor or from the reactor cavity volume
- generation of non-condensable gases as a result of the steam-zirconium reaction and release of gases during the corium-concrete interaction.

The basic measures directed to prevention of late containment failure are as follows:

- start-up or restoration of operation of the recirculation mode including the spray system
- containment venting filter is put into operation
- water supply into the containment compartments, primary circuit or on the containment floor.
4.6 Retention of fission products

The most effective way of retention of fission products at severe accidents is the preservation or restoration of leak-tightness of the primary circuit or provision of the containment leak-tightness. If leak-tightness of the primary circuit is not managed to be provided or restored, then the second barrier on the way of the activity release from the fuel is water. Providing that water for the fuel cooling is supplied sufficiently, the greater portion of radioactive fission products is retained in water. The main portion of those products stay in the reactor vessel, if its integrity being maintained. A portion of products will release with radioactive gases or in aerosol form and a portion of fission products will enter the containment floor together with water.

In the design two versions are considered for cooling the melt fallen from the reactor. In the first version cooling of the corium will occur in the core catcher submerged in the water. In the second version cooling of the melt will happen by spreading the melt over the floor of the reactor cavity having the heat-resistant concrete lining wherein water is preliminary discharged from the reactor internals inspection wells.

The main mass of solid fission products will deposit on the reactor cavity bottom with the time being. Gaseous components will release into the containment air and liquid ones with water will circulate over the all containment rooms.

During functioning of the spray system, the main mass of fission products will be bond up with water with the time being which will effectively retain them.

The use of the containment ventilation permits to decrease the release of radioactive fission products into the atmosphere. Filtered venting of radioactive products via the discharge filter of the containment at pressure increase in it up to 0.7 MPa is performed so as to provide non-exceeding of the limited radioactive release.

Monitoring of radioactive release may be conducted as follows:
- use of gamma-ionization chamber behind the discharge containment filter
- use of the sampling system for analysis of activity in the containment
- radiation monitoring systems for the NPP site and off-site.

Thus, the basic design and engineering measures directed to retention and binding of fission products are reduced to the following:
- maintaining or restoration of leak-tightness of the primary circuit
- maintaining of the vessel integrity
- maintaining of the reactor cavity integrity
- water supply from any systems into the reactor vessel, cavity and containment for cooling and binding radioactive fission products
- irrigation of the containment compartments by the spray system
- assurance of the containment leak-tightness
- assurance of ventilation of the containment and rooms important for the management of severe accident.
5 CONCLUSION

The final step is to develop the Severe Accident Management (SAM) plan, which includes operating guidelines and procedures, necessary hardware and instrumentation. Naturally, these plans must comply with all the above mentioned challenges.

6 REFERENCES

SESSION III

UNCERTAINTIES AND OPEN ISSUES
From Left: Ivan Catton, Mario Bonaca, and John Opeka
OVERVIEW: UNCERTAINTIES REMAINING IN SEVERE ACCIDENT PHENOMENOLOGY

R. E. Henry
Fauske & Associates, Inc.

ABSTRACT

Severe accidents have been considered since the beginning of commercial nuclear power. Initially, only the possible consequences of a severe accident were assessed. With the increased understanding derived through critical thought, fundamental experiments, integral experiments and industry experience, this has evolved into a discipline where issues are considered on design specific and accident sequence specific bases.

Fundamentally there are two reasons for pursuing an understanding of severe accidents.

1. To evaluate the response of a given reactor/containment design for a spectrum of severe accident conditions to determine whether design modifications are warranted. This could be for either an operating plant or a future design and the modifications could relate to accident prevention and/or mitigation of the consequences.

2. To understand the nature of severe accidents, how they progress and how they can be stopped. Such knowledge is generally used to determine the most effective actions for an accident situation as well as to understand what actions are sufficient.

With the above areas, it is important to periodically step back from the phenomenological details and examine where the nuclear community (the industry and the regulators) stand with respect to the two major uses of this knowledge. Only through such periodic assessments can the community focus the limited resources towards addressing the remaining issues. This is one individual’s assessment of where we stand.

1.0 BACKGROUND

Nuclear safety assessments began with the WASH-740 report in 1957 (AEC, 1957). As is commonly known, this study considered the hypothetical consequences of a nuclear accident and had no concept of the reactor design, the accident sequence, or the containment building. WASH 1400 (NRC, 1975), the Reactor Safety Study, was a more indepth investigation which addressed both BWR and PWR designs, the containments associated with the reference designs and a spectrum of accident sequences. This investigation demonstrated that severe accident understanding must consider the specific Nuclear Steam Supply System (NSSS) design, the specific containment design and a spectrum of accident sequences since these influence the system operation, the timing of major events and corrective operator actions. A fundamental conclusion arising out of the reactor safety study was that the Design Basis Accident (DBA)
concept, while a powerful concept in developing designs with substantial margin, does not address all of the issues related to severe accidents. In fact, this investigation concluded that other accidents were more likely and potentially more challenging to the reactor system and containment. Needless to say, the conclusions of the reactor safety study were controversial and in some areas highly criticized and discredited (Lewis, et al., 1978).

As the WASH-1400 controversy quieted, the accident happened at Three Mile Island Unit 2. Many of the accident attributes had been described in WASH 1400. For example, it was a small break LOCA and not a DBA event, the interactions between the operator training, the automatic systems and the instrumentation resulted in the confusion with respect to what was happening in the reactor coolant system (RCS). Once it was clear that an accident was occurring there was further confusion on what actions to take. Public confusion increased as the media speculated on phenomenological issues such as in-vessel steam explosions, secondary recriticalities, hydrogen burns within the reactor coolant system, etc. with no understanding of what they were describing. Hardly the environment for rational thought.

After the TMI-2 event, the NRC raised concerns with possible severe accidents at high population sites, specifically Zion, Indian Point and Limerick. As a format for rationally evaluating the risk associated with these sites, the utilities operating each site performed a plant specific probabilistic risk assessment (CECO, 1981; PASNY/ConEd, 1982; PECO, 1982). With these plant specific studies came the assessment of physical processes on a plant specific basis. This was the first time that attention was devoted to plant specific design features that could influence phenomena such as steam explosions, in-vessel debris cooling, reactor pressure vessel (RPV) failure, high pressure melt ejection (HPME), and ex-vessel debris coolability. In the absence of an analytical model that would express the progression of phenomena, these studies used containment event trees to describe the interaction of physical phenomena on a plant specific and sequence specific basis. Such considerations led the way to more rational thought, experiments and analyses on the mutual influence of these phenomena. These studies were followed by other plant specific evaluations and the industry sponsored IDCOR program directed at providing an adequate technical basis for assessing severe accident behavior.

Entering into integral assessments of severe accidents for different designs was not without pain. Considerations of sequence specific behaviors led to the conclusion that a reactor vessel may fail while at substantial pressure. As a result, considerations were raised with respect to whether the debris could be transported within the containment and the subsequent consequences. This led to considerations of Direct Containment Heating (DCH) for the containment, thereby raising a new issue that must be addressed by the NRC since this was speculated as potentially challenging containment integrity shortly after vessel failure. For BWRs, a similar issue arose with respect to the possibly failure of a Mark I liner as debris would be discharged from the reactor vessel (NRC, 1985). In this evaluation, NRC contractors concluded that molten core debris discharged from the reactor vessel could flow out of the pedestal region, across the drywell floor, contact the containment liner and melt through this liner into the gap between the liner and the biological shield. This was speculated to then cause a release of fission products to the environment. Furthermore, it was concluded that this thermal attack could occur even if there was substantial water on the drywell floor.
At the conclusion of the IDCOR program, the industry assessment was that the likelihood of a severe accident at U.S. nuclear plants was low, the public risk associated with the operation of such plants was low and that the specific risk associated with the given plant was dependent upon the plant design, its operation and the training of the operating staff. In 1988, the NRC issued Generic Letter 88-20 requesting all operators of nuclear power plants to perform an individual plant examination searching for vulnerabilities to severe accidents (NRC, 1988). In this generic letter the operators were requested to assess both the likelihood that an accident would occur and the severe accident issues that would specifically affect the containment performance; sometimes designated as the back-end analysis. Through this assessment, individual utilities evaluated severe accident phenomena for their specific design(s).

In some cases, EPRI and/or individual utilities performed their own experiments to improve the understanding of specific phenomena. For example, challenges to the Mark I liner integrity after RPV failure (Malinovic, et al., 1989), DCH (Henry, et al., 1991) and RPV external cooling (Henry, et al., 1993). In addition, the NRC initiated its own update of severe accident risk using five reference plants (NRC, 1990). In NUREG-1150, the NRC and contractor laboratories concluded that the likelihood of having a severe accident at U.S. plants was low and that the risk to public health and safety associated with severe accidents was also low in comparison with other risks to which the public is exposed. Furthermore, NUREG-1150 concluded that the NSSS and the containment design would have a substantial influence on many of the phenomena.

Lastly, the nuclear industry, through EPRI initiated a study to provide the technical basis for developing severe accident management guidelines (SAMGs) (EPRI, 1992). As part of this Technical Basis Report (TBR), the status of the severe accident phenomena knowledge base, and its relationship to Accident Management (AM) considerations were summarized. Where appropriate the summaries of each phenomenon was integrated with a set of candidate high level actions that could be used to recover from a severe accident. This provided a perspective on how a specific phenomena would be influenced by one or more of these actions. An update of this assessment is given at this meeting (Cheval, 1995). Subsequent to the TBR, significant scale experiments have been performed to further investigate some of the phenomena addressed in previous studies. These are considered in the next section.

2.0 WHERE DO WE STAND AND WHY?

As briefly outlined in Section 1, various phenomena have been considered in different severe accident evaluations. In numerous cases, these phenomena have been considered as being of sufficient importance that they substantially influenced entire evaluations. Table 1 provides a list of some of the phenomena that have influenced particular studies. As noted in this table, several of these phenomena have either been resolved, or are approaching resolution. In this section we will briefly consider these and how resolution has been achieved or is being structured. To provide an assessment on physical phenomena, we first need to develop a list of pertinent phenomena for severe accident analyses.
<table>
<thead>
<tr>
<th>Phenomena</th>
<th>Studies</th>
<th>Year</th>
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<tbody>
<tr>
<td>In-vessel steam explosions.*</td>
<td>WASH-1400</td>
<td>1975</td>
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<td>Debris coolability.</td>
<td>Zion Study</td>
<td>1981</td>
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<tr>
<td>Debris dispersal (DCH).*</td>
<td>Zion Study</td>
<td>1981</td>
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<tr>
<td>In-vessel natural circulation.*</td>
<td>EPRI/IDCOR</td>
<td>1985</td>
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<tr>
<td>Mark I liner.*</td>
<td>IDCOR/NRC</td>
<td>1988</td>
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<td></td>
<td>NUREG-11500</td>
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<tr>
<td>Hot leg creep rupture.*</td>
<td>EPRI/NRC</td>
<td>1988</td>
</tr>
<tr>
<td>External RPV cooling.*</td>
<td>FAI/CECo/UCSB</td>
<td>1989</td>
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<tr>
<td>Steam inverting.</td>
<td>IDCOR/EPRI/NRC</td>
<td>1988</td>
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<tr>
<td>Ex-vessel cooling.</td>
<td>EPRI/NRC</td>
<td>1988</td>
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<tr>
<td>In-vessel cooling.</td>
<td>FAI/INEL</td>
<td>1993</td>
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* Resolved or approaching resolution.

It is well known that fission products can be released from the fuel matrix as a result of overheating that would occur during an accident. As a defense against this, the containment is designed to be isolated and contain essentially all the fission products. Hence, the uncertainties associated with fission product behavior are of little concern as long as the containment remains isolated, which depends on the thermal hydraulic behavior associated with the accident progression and recovery from the accident state. Therefore, here we focus on the uncertainties associated with thermal hydraulic issues in severe accidents. Table 2 is a general categorization of thermal hydraulic phenomena. Several of these are a combination of other smaller phenomena, but this characterization serves the purpose for this evaluation.
Table 2
Severe Accident Physical Phenomena

<table>
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<tr>
<th>Phenomena</th>
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<tr>
<td>1. Clad oxidation.</td>
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<td>2. Core melt relocation.</td>
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<tr>
<td>3. Molten pool in core.</td>
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<tr>
<td>5. RCS failure modes.</td>
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<tr>
<td>6. In-vessel steam explosion.</td>
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<tr>
<td>7. In-vessel steam generation.</td>
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<tr>
<td>8. In-vessel debris formation.</td>
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<tr>
<td>9. RPV failure modes.</td>
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<tr>
<td>10. In-vessel cooling mechanism(s).</td>
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<tr>
<td>11. RPV external cooling.</td>
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<tr>
<td>12. Ex-vessel steam explosion.</td>
</tr>
<tr>
<td>13. Direct containment heating.</td>
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<tr>
<td>14. Mark I liner attack.</td>
</tr>
<tr>
<td>15. Ex-vessel debris cooling.</td>
</tr>
<tr>
<td>16. Steam inerting of the containment.</td>
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<tr>
<td>17. Hydrogen burning in containment.</td>
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</table>

Clad Oxidation, Core Melt Relocation, Molten Pool and Crust Behavior

These have an extensive basis from experiments at various scales, the TMI-2 experience and numerous system calculations using RELAP/SCDAP and the MAAP codes. Assessments for a large variety of plants and accident sequences show that the extent of hydrogen produced generally lies within the range of 30% to 75% of the active cladding being oxidized. Hence, the directions given by the NRC following the TMI-2 accident with respect to the extent of cladding oxidation are still appropriate. Furthermore, the experience in the TMI-2 vessel investigation project (Wolf and Rempe, 1993), as well as the integral investigations of RELAP/SCDAP and the MAAP4 codes is that much of the metallic zircaloy relocates to the lower regions of the reactor core and the material which drains into the RPV lower head is
principally oxidic, i.e. $\text{UO}_2$ and $\text{ZrO}_2$. This suggests that a substantial fraction of the zirconium would result in a blockage of the lower core region with little oxidation potential and, while this could be oxidized during ex-vessel core-concrete attack, it would not be substantially oxidized in the RCS. This further supports the direction given by the NRC that individual designs must consider the oxidation of 75% of the active cladding for severe accident conditions.

RCS Failure Modes

RCS failure modes have been a relatively recent addition to the list of phenomena and have evolved from the considerations of natural circulation within the primary system. In particular, this has focused on the potential failure of hot legs in PWR designs for conditions where the reactor core is uncovered for a substantial interval. Overheating of the hot legs, and potentially the steam generator tubes for the inverted U-tube designs, has been the subject of a detailed scaled experiment (Stewart, et al., 1986) and numerous analyses with the NRC (RELAP/SCDAP) and industry (MAAP4) integral system codes. All of these conclude that natural circulation would be established between an overheated core and the upper plenum region and additional natural circulation circuits would result in the hot legs and within the steam generator tubes. Furthermore, if injection to the reactor coolant system is not recovered, creep rupture of one of the hot legs (or perhaps the surge line) would occur well before the steam generator tube integrity would be challenged. Given the substantial differences in approach taken by the analyses and the existence of scaled experiments, there is sufficient information available to conclude that this material creep behavior would occur and this should be considered in developing the SAMGs. Hydrogen releases from the RCS depend on where the RCS fails and other phenomena such as High Pressure Melt Ejection (HPME), depend on whether the RCS failure occurs.

In-Vessel Steam Generation, Steam Explosions and Debris Formation

With the TMI-2 experience it is clear that debris drainage into the RPV lower plenum must be considered even when the damaged core is completely submerged in water. It was also clear that no explosive interaction was created in the accident, which has been attributed to the elevated RCS pressure. Substantial works have been reported on the influence of pressure to suppress explosive interactions (Henry and Fauske, 1979; Hohmann, et al., 1979; Hohmann, et al., 1982). Recent experiments performed in the FARO facility (Magallon and Hohmann, 1993) and the ALPHA test in Japan (Yamano, et al., 1993) have further supported the influence of elevated pressure with the latter experiments showing that a pressure of 1.6 MPa is sufficient to suppress an explosive interaction. (This information is consistent with that presented by the previous experimental programs.)

Other experimental and analytical studies have focused on the potential for fine scale particulation and mixing of large quantities of high temperature melt and water (Angelini, et al., 1993 and Fletcher and Denham, 1993). Both investigations concluded that it is extremely difficult to mix large quantities of high temperature melt with water and also that substantial steam is formed during the premixing which depletes the water in the interaction zone. This is a more refined assessment of the mechanism proposed by Henry and Fauske (1981) which suggests that vapor formation during the premixing limits the molten material involved.
In 1985, the NRC formed the Steam Explosion Review Group (SERG) (NRC, 1985) and chartered this group to assess the likelihood of $\alpha$-mode failure. The consensus was that explosive interactions sufficient to rupture the primary system and therefore the containment were very unlikely. During the 1993 CSNI-FCI Special Meeting in Santa Barbara, similar questions were asked, and again, the consensus was that the work performed since the 1985 meeting supported the conclusions made by the SERG. In many instances, additional work had further refined key arguments related to the inability to establish the necessary initial conditions. In June of 1995, the NRC is sponsoring a workshop to update the understanding with respect to steam explosions. With the additional experiments provided in the FARO (Magallon and Hohmann, 1993) and ALPHA facilities, as well as the additional analyses performed by Angelini, et al. (1993) as well as Fletcher and Denham (1993) it would appear that there is a developing consensus on the $\alpha$-mode failure issue.

Debris particulation and in-vessel steam generation are also part of the understanding related to steam explosions. In particular, significant work has been performed with respect to the breakup of molten jets as they pour through water (Burger, et al., 1993). Moreover, the FARO experiments provide a substantial scale, real material demonstration of the steaming rate during this process. From the information accumulated to date, including the TMI-2 Vessel Inspection Project (VIP) (Wolf and Rempe, 1993), some particulation may occur, but the assessment of debris within the lower plenum must also consider that there is a substantial material layer which does not particulate. Particulate debris causes a net steam generation to the RCS with some potential for additional pressurization as was observed in the TMI-2 accident. These integral system details are part of the lower plenum modeling in the MAAP4 code (EPRI, 1994). With the extensive information available on debris particulation and the net steaming rate from molten material draining into the lower plenum, the major issue is how much material is not particulated since this results in a potential threat to the RPV wall integrity.

**RPV Failure Modes**

RPV failure modes were initially considered in the Zion Probabilistic Safety Study (CECo, 1981). In this study, molten core debris draining into the lower plenum was postulated to challenge the limited depth welds that anchor the in-core penetrations in the RPV. Similar considerations were used for the BWR evaluation of the Limerick plant (PECo, 1982). As the core removal progressed in the TMI-2 vessel and more was learned of the accident scenario, including the drainage of molten core material into the lower plenum, it became apparent that these welds were more robust than was previously credited. Since this is a generic issue, the details of how the vessel would fail are not particularly important in the IPEs which searched for plant specific vulnerabilities. However, for AM evaluations, the time available to recover from an accident and keep the core debris in the RPV, become of key importance. In particular, retaining debris within the RPV lower head eliminates the uncertainties associated with ex-vessel debris behavior and therefore minimizes the uncertainties to be considered by the AM team. EPRI sponsored full scale experiments on the challenge to in-core penetrations by molten material, with particular emphasis on the possibility that molten debris could flow through the central passage used by the traveling in-core probe (TIP) (Hammersley and Henry, 1994). These experiments revealed that the in-core penetrations experienced molten debris traveling through the TIP passage but that this material quickly froze and plugged this flow path so completely that
there was no further depressurization of the simulated RCS. Moreover, there was no challenge to the supporting weld for the penetration and analyses indicated that these welds would have to be essentially melted before the penetration could be ejected. Hence, these experiments provided the fundamental insights to create mechanistic models for the MAAP code to represent this lower plenum behavior. These models show that the penetration behavior is far removed from a failure condition when molten debris drains into the lower head, even if the RCS pressure is at the nominal operating values.

Other experiments were performed on the 5 cm (2 in.) water filled drain lines that are in some BWR vessels. Two experiments were performed and there was no indication of significant strain in the drain line even though the entire line was filled with molten oxidic material. In fact, these experiments showed a considerable potential for creating a significant contact resistance between the debris and the drain line which influenced the energy transfer from the molten material to the wall. Other experiments were carried out with molten material draining into a dry lower plenum without penetrations and also with water in the lower plenum. These tests also demonstrated the formation of an interfacial contact resistance if the molten debris drains through water. This was the foundation for this model in the MAAP4 code and also provided insights into the possibility of in-vessel cooling that is discussed below.

As a result of the experiments on RPV failure modes, it is clear that the lower head penetrations are not nearly as susceptible to failure as was once considered. In fact, there appears to be virtually no potential for failing a penetration immediately after core debris would drain into the lower head. If the core debris accumulates in the lower head, dries out, heats the reactor vessel wall sufficient to cause extensive strain of the wall, then there is a potential that the penetrations could be the failure site. This mostly depends on the accident sequence and is certainly dependent on whether water has been added to the reactor coolant system or whether external cooling of the RPV is developed. There is a potential for a much higher heat flux towards the equator of the hemisphere, however, the accident sequence generally dictates that molten debris arrives in the bottom of the reactor vessel long before the melt accumulates sufficiently to approach filling the RPV lower head. As a result, there are uncertainties in where vessel failure would occur if the RPV head is dry, but it is clear that there would be no immediate failure as a result of debris entering the lower plenum.

**In-Vessel Cooling**

As a part of the TMI-2 core removal, there was some excellent detective work performed on the lower head, to assess the thermal response. Consideration of this thermal-mechanical response has resulted in a proposed mechanism (Henry and Dube, 1994) for in-vessel cooling which characterizes the cooling of the RPV wall with water that is in the RCS. In this mechanism, limited wall strain, of the order of a few hundred microns, is sufficient to enable water to ingress between the RPV wall and the debris to cool the wall and prevent further strain, and therefore RPV failure. Hence, there is important feedback between the core overheating the RPV wall, straining of the wall and water ingress into the small gap between the debris and the wall. It is important to note that the extent of strain is very small compared to that which would potentially threaten the RPV integrity. It is also important to note that the RPV lower head will not fail without substantial strain. This model has been added to the MAAP
lower plenum models including material creep in the presence of a strong radial temperature gradient in the carbon steel wall. Benchmarking with the TMI-2 accident gives a consistent picture with molten debris draining into the lower head, substantial overheating of the wall and cooling of the RPV wall when limited strain occurs. While there are other possible explanations, it is important to note that substantial strain must occur before failure and integral code calculations must consider such strain, and its implications, before assessing the potential for RPV failure.

This is a relatively new mechanism but it also has substantial importance to the accident management assessments because there is an obvious benefit to keeping core debris within the reactor vessel. This can be done by either cooling the debris within the RPV, external cooling of the RPV, or both. Certainly adding water to the RCS is the appropriate action when injection is available. For many reactor containments, external cooling is accomplished in a relatively easy manner. On the other hand, there are also numerous designs in which it is difficult to flood the containment and it may also be difficult to cool the RPV lower head because of a vessel support skirt. In these cases, flooding the containment under conditions in which the ECCS injection has been restored may not be an appropriate action. Containment flooding is difficult and in many cases eliminates some of the important features, such as pressure suppression for the Mark I and Mark II designs, as well as potentially flooding key instruments. Therefore while in-vessel cooling can not be considered as resolved because it is so new, it is important to perform the necessary work to understand this cooling potential, because of the importance to accident management decisions.

**RPV External Cooling**

External cooling was discussed extensively in the TBR, both in terms of the physical processes involved and the design specific features that could limit this cooling. At the time, there was only limited experimental data, but this data showed a substantial margin between the surface heat removal required to maintain RPV integrity and that which would cause a boiling crises. Since then additional experimental works have been reported to further support that external cooling is a viable mechanism for preventing RPV failure as long as water has access to the RPV wall. In particular, the two dimensional critical heat flux experiments by Theofanous, et al. (1994) showed the heat removal capabilities to be far in excess of the energy transfer that would be expected under accident conditions. In addition large scale three-dimensional experiments were performed at Sandia National Laboratory on a vessel with an elliptical lower head. With this particular head design, the larger radius of curvature provided a conservative representation of the capabilities for boiling heat transfer on the downward facing surface (Chu, et al., 1994a and Chu, et al., 1994b). Both of these experiments further supported the capabilities for RPV external cooling. Hammersley, et al. (1993) have performed experiments for vessels with support skirts and have shown that this can preclude water from contacting the RPV lower head. With the various experiments that have been performed, this issue can be considered as resolved for accident management behavior. Of course the plant specific features, such as a vessel support skirt, must be considered before external cooling is credited.
Direct Containment Heating and Ex-Vessel Steam Explosions

These are considered together since much of the experimental work on ex-vessel steam explosions comes from the DCH experiments. In particular, a number of experiments have been performed with water in the simulated reactor cavity. These have measured significant dynamic pressurizations of the reactor cavity but these interactions have not led to any other consequences other than expelling the debris from the reactor cavity and producing steam that could oxidize metal in the molten debris.

As part of the Zion IPE study, Commonwealth Edison sponsored experiments to investigate the DCH potential in a 1/20th linear scale representation of the Zion containment (Henry, et al., 1991). At about the same time the NRC initiated a scaling methodology committee, chaired by Dr. Novak Zuber, to address scaling issues for severe accidents using direct containment heating to both develop the methodology and demonstrate its usefulness (Zuber, 1993). Linear scaling was judged as the most appropriate means of addressing DCH and the NRC initiated programs at two different scales at Argonne National Laboratory (1/40th) and Sandia (1/10th) for Zion-like systems. These tests provided an excellent application of the proposed methodology and experimentally provided direct insight into scale related issues. These experiments (Binder, et al., 1994 and Allen, et al., 1994) showed remarkable similarity in the containment dynamic behavior for counterpart experiments. Hence, these demonstrated confidence in the scaling methodology as well as that the containment pressurization exhibited was much less than that which would challenge the containment integrity. Specifically, the net pressure increase for a substantial inventory of melt ejected from the cavity was approximately 1 bar for an inerted containment and 2.5 bars if hydrogen combustion occurred. Further it was noted that pre-existing hydrogen in the containment atmosphere was not burned on a timescale that was meaningful with respect to the containment pressurization. Hence, the hydrogen that was burned during the pressurization was that created by oxidizing high temperature metals during the transport through the reactor cavity and the steam generator region. With the development of the scaling methodology and the successful completion of the Zion-like experiments at two different scales and the resulting pressure increases that are much less than those that would challenge the containment integrity, this issue has been closed and documented in Pilch, et al. (1994).

A similar set of experiments were performed at 1/10th and 1/6th scale mockups of the Surry containment in facilities at the Sandia National Laboratory (Blanchat, et al., 1994). Containment compartmentalization had a substantial influence on the pressurization that could occur within the containment as a result of high pressure melt ejection. Here again the pressure increase in containment was approximately 2.5 bars for conditions in which hydrogen combustion could occur but the hydrogen burned was essentially created during the event. Hydrogen that was previously in the containment atmosphere did not appear to be consumed on a timescale that significantly influenced containment pressurization. With the successful completion of the test program at the two different scales and the examination of the conditions in the RCS that could result in depressurization due to hot leg creep rupture, the DCH issue for Surry-like containments has also been closed. This has recently been documented for the NRC by Sandia Laboratory personnel (Pilch, et al., 1995).
The IDCOR program (IDCOR, 1985) identified several types of reactor cavity configurations. Because of these variations, some additional experiments are underway at Sandia National Laboratory to address these differences. In particular, the extensive work done on issues related to hot leg creep rupture have demonstrated that an uncovered reactor core would lead to hot leg creep rupture and depressurization of the reactor coolant system before any high temperature molten material would drain into the lower plenum. Consequently, the only conditions that would be considered as realistic for an HPME event would be those in which the reactor core would be covered by water, as was the case during the TMI-2 accident. Hence, for these conditions high temperature core debris would be pushed out of the reactor coolant system by saturated water. Note that this set of conditions conflicts with the in-vessel cooling mechanism that suggests that vessel failure would not occur if water were present within the reactor system well before vessel strain were to occur. Hence, this set of conditions may not exist, but are being considered a part of closure for the DCH issue.

**Mark I Liner Attack**

This issue was raised by the NRC Containment Loads Working Group (NRC, 1985) and was addressed by significant scale experiments (Malinovic, et al., 1989). Issue resolution was addressed by the NRC through a structured analytical and experimental program (Theofanous, et al., 1990). In this approach, a general characterization of the conditions which could possibly challenge the Mark I liner were formulated and evaluated in a probabilistic structure using Peach Bottom as a reference plant. In the final analysis, the assessment concludes that if there is no water in the drywell at the time that core debris is released from the reactor vessel, there is a high probability that the containment liner could be attacked. Conversely, if there is water in the drywell at this time, attack of the Mark I liner is not physically credible. This approach to resolution of the Mark I liner issue was submitted to a peer view committee. Numerous comments were provided to the authors and additional, separate analyses were performed to evaluate and address the various comments. Eventually technical closure was developed on all comments and the Mark I containment liner issue is resolved.

**Ex-Vessel Cooling**

Ex-vessel debris cooling has been the subject of substantial discussion since the Zion Probabilistic Safety Study. Experiments have been performed, such as those discussed above for the Mark I liner attack, to obtain a perspective on the rate of cooling resulting from water ingestion. There are substantial scale issues associated with ex-vessel cooling, including the properties of the core debris, how these physical properties could be altered as molten concrete is added to the debris as thermal attack progresses. Significant scale experiments have been performed and are discussed in the literature (Epstein, 1992) which demonstrate the possibility that there is a significant heat removal from core material in the ex-vessel mode. However, these studies have not been able to clearly identify a long term cooling mode. Furthermore, performing such experimental investigations are difficult at best, and must be performed at a significant scale, which is judged to be at least 1 meter by 1 meter, i.e. substantial quantities of uranium dioxide.
Because of the experimental difficulties and the substantial scale related issues associated with water ingestion, the details of ex-vessel coolability may never be clearly understood. On the other hand, with respect to accident management actions, a detailed understanding may not be required. In all cases, the discharge of core debris from the RCS into containment requires that the debris be submerged in water for several reasons. Firstly, water can cool the debris by ingestion. Secondly, submerging the core debris would scrub fission products that could be released as a result of core-concrete attack even if the debris were not coolable. Lastly, submerging the debris eliminates any substantial heat load from the core material to other containment structures that are potentially sensitive to elevated temperatures. For all these reasons, it is clear that water should be added to the containment if it is suspected that core debris has been lost from the RPV, regardless of the cooling capacity of the water itself. Hence, while a more detailed perspective of ex-vessel cooling is desired, there is enough known from current information to support the necessary decision making for accident management.

**Hydrogen Burning and Steam Inverting in the Containment Atmosphere**

These two are combined for obvious reasons. Substantial investigations have been performed, as well as large scale experiments (Thompson, et al., 1988a and Thompson, et al., 1988b), and with these, the importance of hydrogen combustion in the atmosphere is understood as are the influences of accident consequences such as steam inverting. It is important that AM decision making appreciates the possibility of de-inverting the containment atmosphere as a result of a candidate high level action, i.e. containment sprays. In particular, containment sprays both de-inert the atmosphere and increase the turbulence which can increase the burning rate (Thompson, et al., 1988a). Also, the rate at which the atmosphere is de-inerted can be important since the retention of a significant steam partial pressure tends to limit the burning rate. Thus, while this is important for accident management evaluations, the existing knowledge base appears to be sufficient.

In summary, most of the elements related to the understanding of severe accidents have a sufficient knowledge base. Table 3 summarizes the above discussions on whether these major phenomena are sufficiently understood to be considered as resolved, whether they are approaching resolution, or whether more work is needed to achieve the necessary understanding to support AM decision making.

**3.0 WHAT IS LEFT TO DO?**

As indicated in Table 3, the understanding for most phenomena is sufficient to support accident management decision making, and more specifically the development of SAMGs. Only in limited cases are there areas where this understanding could be further refined and these principally relate to stopping the accident progression by cooling the core debris. The focus of these activities can be determined by considering the key events for an accident sequence and why they are so important. As demonstrated, the last two focus on stopping the accident sequence and preventing the release of fission products from the containment. By keeping the debris within the RPV, the uncertainties associated with ex-vessel cooling and other containment issues are essentially eliminated.
Cooling within the RPV can be done by either water injected to the RCS, external RPV cooling, or both. While each has its own uncertainties, those associated with external cooling are substantially less because of the larger experimental database. However, some reactor designs do not permit effective cooling of the RPV lower head and for others, achieving this state is very difficult. Therefore, to provide a necessary basis for accident management decision making, two areas would be particularly helpful to achieve closure. The first is to provide the technical depth to the in-vessel cooling phenomena; specifically to create the necessary level of confidence that is required to clearly define when sufficient actions have been taken for cooling the core material. Secondly, the issues associated with external RPV cooling have only addressed those related to heat removal from the RPV lower head. Additional experiments showing the removal from the remainder of the RCS, including the RPV cylinder and the hot legs, would be beneficial. In particular, this could help address the issue of: if only external cooling was available, what water level would be sufficient to stop the accident progression?

As indicated by the above discussion, those issues necessary to support accident management decision making are rapidly approaching closure. Once technical closure has been achieved on the list of phenomena given in Table 2, the only remaining element is to establish effective training for the operating and technical staffs. This training should not be presented in a burdensome manner, but should be done in a streamlined fashion, which identifies the state of knowledge of all the major phenomena and how these are addressed. It is important that this training be reinforced on a regular basis to (1) demonstrate that this is a living process which continues to take advantage of an increasing knowledge base from scientific studies, integral analyses and industry experience and (2) ensure that the influence of these physical processes is continually reviewed with the operating and technical staffs. In this regard, the structure of physical phenomena as discussed in this paper and as applied to the four key events would be an effective approach to cover both the basics of accident management as well as to touch on the more esoteric elements of individual phenomenon. With this background and training, the operating and technical staffs have the wherewithal to address the two reasons for understanding severe accidents:

1. to determine whether design modifications are warranted, and

2. to understand the nature of severe accidents, how they progress and how they can be stopped.
<table>
<thead>
<tr>
<th>Phenomena</th>
<th>Resolved</th>
<th>Virtually Resolved</th>
<th>More Work Is Needed</th>
</tr>
</thead>
<tbody>
<tr>
<td>1. Clad oxidation.</td>
<td>✓</td>
<td></td>
<td></td>
</tr>
<tr>
<td>2. Core melt relocation.</td>
<td>✓</td>
<td></td>
<td></td>
</tr>
<tr>
<td>3. Molten pool in core.</td>
<td>✓</td>
<td></td>
<td></td>
</tr>
<tr>
<td>5. RCS failure modes.</td>
<td>✓</td>
<td></td>
<td></td>
</tr>
<tr>
<td>6. In-vessel steam explosion.</td>
<td>✓</td>
<td></td>
<td></td>
</tr>
<tr>
<td>7. In-vessel steam generation.</td>
<td>✓</td>
<td></td>
<td></td>
</tr>
<tr>
<td>8. In-vessel debris formation.</td>
<td>✓</td>
<td></td>
<td></td>
</tr>
<tr>
<td>9. RPV failure modes.</td>
<td>✓</td>
<td>✓</td>
<td>✓</td>
</tr>
<tr>
<td>10. In-vessel cooling mechanism(s).</td>
<td>✓</td>
<td></td>
<td>✓</td>
</tr>
<tr>
<td>11. RPV external cooling.</td>
<td>✓</td>
<td></td>
<td></td>
</tr>
<tr>
<td>12. Ex-vessel steam explosion.</td>
<td>✓</td>
<td></td>
<td></td>
</tr>
<tr>
<td>13. Direct containment heating.</td>
<td>✓</td>
<td></td>
<td></td>
</tr>
<tr>
<td>14. Mark I liner attack.</td>
<td>✓</td>
<td></td>
<td></td>
</tr>
<tr>
<td>15. Ex-vessel debris cooling.</td>
<td>?*</td>
<td>?*</td>
<td>?*</td>
</tr>
<tr>
<td>16. Steam inerting of the containment.</td>
<td>✓</td>
<td></td>
<td></td>
</tr>
<tr>
<td>17. Hydrogen burning in containment.</td>
<td>✓</td>
<td></td>
<td></td>
</tr>
</tbody>
</table>

* Enough is likely known for AM evaluations even though substantial uncertainties remain.
<table>
<thead>
<tr>
<th>Event</th>
<th>Increase in Accident Severity (Consequence)</th>
</tr>
</thead>
<tbody>
<tr>
<td>Core is Uncovered</td>
<td>Core integrity is challenged.</td>
</tr>
<tr>
<td>Major Core Damage</td>
<td>Fission products released to the containment.</td>
</tr>
<tr>
<td>RPV Failure</td>
<td>Core debris is discharged to the containment.</td>
</tr>
<tr>
<td>Containment Failure</td>
<td>Fission products released to the environment.</td>
</tr>
</tbody>
</table>

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Reactor Cavity Flooding as an Accident Management Strategy

by

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ABSTRACT

This study deals with the uncertainty in the parameters of deterministic models important to flooding the reactor cavity as an accident management strategy. Among the parameters, a few that can significantly affect the output of the model were selected, based on several preliminary calculations (sensitivity study for each parameter) and physics. For the selected parameters, probabilistic distributions which represent lack of knowledge were specified. It was found that when the reactor cavity is flooded to cool the vessel lower head, the lower head shell may not melt through and fail due to stress. Creep may be the only possible failure mechanism. When the RCS is at high pressure, vessel lower head failure is highly probable but the time of the failure is uncertain due to uncertainty in the parameters in the model. If AC power is recovered before vessel failure, the RCS can be depressurized and the vessel lower head may be stable. A further study of the availability of a system for depressurization at AC power recovery timing is needed. When the RCS is at low pressure, there is no load to cause creep. The vessel lower head would be stable.

I. INTRODUCTION

Nuclear power plants require well defined operational procedures for accident conditions as well as for normal conditions in order to operate the systems efficiently and safely. Emergency operating procedures are well defined for both pressurized and boiling water reactors (PWRs and BWRs) up to the time of inadequate core cooling, but have not been fully developed for severe accident conditions involving significant core damage.

Many candidate strategies for managing PWR and BWR severe accidents were identified and discussed during two workshops held at the University of California, Los Angeles [Kastenberg et al., 1991]. For each candidate strategy, uncertainties exist and involve issues related to phenomena, operator actions, instrumentation, and system availability. In particular, large uncertainty in phenomena occurs because operator actions change the progression of a severe accident, and introduce new physical regimes such as temperature or pressure, and new conditions such as the presence or absence of water.

For assessment of severe accident management strategies for nuclear power plants, we have come to rely heavily on expert opinion to develop descriptions of accidents which are not well understood. One reason for doing so is because events of concern to analysts are very rare and experimental information on the basis of which predictions can be made is not easily available. Often the expert opinions are implicitly understood and thereby not explicitly stated. These implicit expert opinions are not questioned when there is widespread agreement [Chhibber et al., 1992]. When there is disagreement and uncertainty, however, expert opinions
need to be explicitly stated to clarify the uncertainty in them.

The purpose of this paper is to demonstrate how one might explicitly represent phenomenological uncertainty when giving an opinion for certain phenomena, given the state of the art in knowledge about the phenomena, and then to illustrate how the uncertainty impacts the assessment of severe accident management strategies. Here, severe accident management is considered as a decision making problem with uncertainty, which can be solved using the principles of probability theory and utility theory. Influence diagrams [Jae and Apostolakis, 1992] are utilized as an analytical tool for assessing the viability of a severe accident management strategy.

Because there are too many accident sequences and nuclear power plant (NPP) types, a NPP and an accident sequence for the NPP must be specified. In this study, a PWR (the Surry station) undergoing a short term station blackout (TMLB') is selected. The selected accident sequence is the dominant internal event leading to core damage in a PWR. For station blackout sequences, a strategy involving lower cavity flooding for lower head cooling has been proposed to keep the molten core and structural debris within the vessel (forms a crucible). This was the strategy selected for study by Kastenberg et al. (1993) at UCLA where the selected strategy was assessed using an influence diagram. Figure 1 is a modified version of the influence diagram developed at UCLA for the Surry station (PWR). In this paper, estimates of the effectiveness of the accident management strategy are improved by better treatment of selected phenomena.

The influence diagram points to two important severe accident phenomena with significant phenomenological uncertainty among the those that occur if the strategy is employed. The probability distribution of a particular outcome will be estimated based on our current understanding of these two phenomena. The strategy is then assessed using the influence diagram shown in Figure 1 with the calculated probability distributions of the selected phenomena. The remaining probabilities needed for the evaluation of the influence diagram are taken from the UCLA study. The effects of uncertainty in the selected phenomena on the results are then studied.

The two selected phenomena are chosen by reviewing the influence diagram. First, the effectiveness of cavity flooding as a strategy for saving the vessel lower head is considered. If cooling from outside the lower head saves the vessel, the strategy is proven to be effective. Second, heatup of the upper part of the primary reactor coolant system (RCS) due to buoyancy driven recirculation is considered. If the RCS pressure boundary remains intact, hot gas in the RCS transfers heat by natural convection from the core material to more vulnerable parts of the RCS such as the steam generator tubes, the pressurizer surge line and the hot leg.

Scope of the Study

Deterministic models simulating vessel lower head failure and buoyancy driven recirculation are used to determine the outcomes of certain events. There exist large parameter and data uncertainties associated with the predictions of the outcomes of the phenomena using the models. The uncertainty in the predictions needs to be explicitly stated. For an uncertainty statement to be complete, the range of all admissible models, and the distribution of parameters that fit those models, should be considered [Chhibber et al., 1991].

The models are not first principle representations of the physics. As a result they contain, or need, specified parameters, e.g., particle sizes, heat transfer coefficients, void
fraction, and friction factors in a steam explosion model [Fletcher, 1991]. There is a lack of knowledge as to what their values should be. This is not due to stochastic variability but due to incomplete understanding of the physics. Although there may be some stochastic variability, most of the uncertainty comes from lack of knowledge.

Among the many parameters, those that can significantly affect the outputs of the model are selected based on preliminary calculations (sensitivity study for each parameter) and physics. For the selected parameters, probabilistic distributions which represent a lack of knowledge are assigned. While certain bounds arise from physical considerations or observations, specification of the parameter distributions usually depends on engineering judgement. The assigned distributions needed to fill the gap between the need-to-know and the lack of data represent one's subjectivity. Parameter uncertainty has been treated by propagating the assigned probabilistic values through the model using random sampling techniques such as Monte Carlo simulation.

In this study, the uncertainty in the parameters of a given model is dealt with. Given a state-of-the-art model, a distribution of output of interest is obtained by making many calculations using a model, i.e., the propagation of uncertainty in parameters through the model. The resulting distribution could be considered as a substantive expert opinion. However, to estimate the distribution, a sufficient number of calculations using the model are not practical because of limited resources (time and money). To perform enough calculations to generate the output distribution, the model is replaced with a response surface generated from several outputs of the model, given systematically selected parameter values covering the possible ranges of their uncertainties. The sensitivity of the output distribution to the uncertainty in the parameters is also investigated by fixing one or more of the parameters and allowing the others to vary according to specified distributions.

Finally, by using a relation between the output of the model and the probability that a certain event occurs (e.g., failure or nonfailure), the probability distribution of the certain event is determined. The determined probability distribution can be interpreted as one's expert opinion with uncertainty. The estimated probability distributions are used to assess the flooding strategy using the influence diagram. The effect of the uncertainty in the parameters on the assessment of the strategy is also presented.

In this study, model uncertainty is not dealt with because quantification of model uncertainty is better left to a normative expert. Model uncertainty occurs due to incomplete understanding or description of the physical phenomena. Several different models often attempt to describe the same phenomenon, each yielding a different result. Efforts to quantify model uncertainty have received growing attention in the past few years. Examples are the work of Apostolakis [1989, 1990, 1993] and of Chhibber et al. [1991, 1992, 1993]. A major thrust of these efforts has been to express degrees of belief in various models. With a lack of complete understanding of the mechanism of a phenomenon, quantification of model uncertainty will be a subjective process. Therefore, subjective probabilities appear to be a natural, intuitive method to express our belief in a model's efficacy or its credibility. This work will not specifically address such issues.

**TWO UNCERTAIN PHENOMENA FOR A PWR**

Two severe accident phenomena that are important when evaluating the effectiveness of lower cavity flooding are selected and estimates of the probability of a particular outcome of
the phenomena are made based on our current understanding of the phenomena. The possibility of vessel lower head failure and upper RCS component failures due to buoyancy driven recirculation are considered. The strategy is then assessed using the influence diagram method with the calculated probabilities. Details about the models and methods used to obtain the vessel lower head and RCS failure probabilities were documented by Lim [1994] and are only summarized here.

Vessel Lower Head Failure for PWR

If the initiated accident sequence leads to loss of core cooling, the core temperature rises and melting starts. The molten core material (corium) slumps and forms a solid debris bed in the vessel lower head because the cavity is flooded to enhance cooling of the vessel lower head. The transient phase from core meltdown to a coolable corium pool configuration in the vessel lower head involves a wide spectrum of phenomena: core melt progression and relocation; core melt fragmentation, quenching and attack against the lower head and penetrations; the dryout and remelting of an in-vessel core debris bed, and the establishment of convective currents in molten corium. Oikkonen (1994) pointed out that a long-term melt retention inside the vessel could not be guaranteed if the vessel lower head did not sustain the transient from core meltdown to a coolable corium pool configuration in the vessel bottom head.

The physical situation can be described as follows; The decay heat generated in the well mixed homogeneous corium pool is convected to the pool boundaries and conducted through the enclosing corium crust to the vessel shell. The heat flux from the pool is dissipated through the vessel shell directly to external boiling water. In the upward direction, the heat flux is dissipated by radiation to the upper portions of the vessel wall. Figure 2 shows the schematic diagram of the PWR vessel lower head where the core material is relocated.

Convective heating of the vessel lower head is a well defined physical problem. Conductive heat transfer governs thermal behavior of the vessel lower head shell. Heat transfer coefficients of the inner and outer surfaces of the shell can be decided with little uncertainty. Mechanical behavior of the shell is relatively complicated but, with some assumptions, can be described by well known mathematical formulations. Most phenomenological uncertainty in this issue exists in initial and boundary conditions needed in the mathematical formulations. Little uncertainty exists in the mathematical formulations of this issue, i.e. governing equations and constitutive relations. Initial and boundary conditions make this issue uncertain. A comparison of the results of 1-D modeling with those of 2-D modeling was made by Lim [1994]. He found that, for the most part, these made little difference.

In this study, the rupture time for the vessel lower head subjected to the given system pressure is determined by using parametric extrapolation techniques. The Larson-Miller relationship (LMP) is evaluated [Park, 1992] to determine the material rupture time as a function of stress and temperature. The applied stress on the shell is assumed to be uniform across the shell. Creep rupture times are determined by carrying out time dependent stress calculations and evaluating the time integrated damage to a particular element in the wall. A damage-based failure criterion is used for estimation of the creep rupture time. Damage is defined using the life-fraction rule [Robinson, 1938] on the basis of rupture time calculated using the LMP.

It was found that creep rupture may be the only possible vessel lower head failure
mechanism under the selected ranges of uncertain parameter values given in Table 1. When the RCS is at low pressure, there is no load to cause creep and the vessel lower head becomes stable. When the RCS is at high pressure, the lower head will fail, but the time of vessel failure is uncertain. Because of the presence of water outside of the lower head, the only failure mode is assumed to be bottom head failure.

The one dimensional models were used to create a response surface by varying the important parameters over appropriate ranges. A Monte Carlo method based on selected parameter distributions was then used to develop the probability of time to vessel failure. Table 2 shows the mean, 5, 50 and 95 percentile values and variances of the vessel failure time under the various conditions. It can readily be seen that the pool volume considerably affects the $t_p$ distributions. It can be concluded that the pool volume is most important in terms of contributing to the uncertainty in $t_p$. The $t_p$ distribution is not sensitive to the decay power because at the time of interest, the decay power is not rapidly reduced and the probable

Table 1. List of Normalized Independent Variables and Calculation Points for PWR.

<table>
<thead>
<tr>
<th>Uncertain Parameters</th>
<th>Calculation Points</th>
</tr>
</thead>
<tbody>
<tr>
<td>$V_p$ (x Core Volume)</td>
<td>Low: 0.2, 0.4, 0.6, 0.8, 1.0</td>
</tr>
<tr>
<td>$Q_p$ (x Full Power)</td>
<td>Low: 0.42%, 0.47%, 0.52%</td>
</tr>
<tr>
<td>$\epsilon_p$</td>
<td>Low: 0.2, 0.35, 0.5, 0.65, 0.8</td>
</tr>
</tbody>
</table>

Table 2 Mean Value and 5, 50, and 95 Percentile Values of Vessel Failure Time, $\log_{10}$ ($t_p$ hours), under Various Conditions.

<table>
<thead>
<tr>
<th>Base Case</th>
<th>Mean</th>
<th>5%</th>
<th>50%</th>
<th>95%</th>
<th>Variance</th>
</tr>
</thead>
<tbody>
<tr>
<td></td>
<td>1.54</td>
<td>-1.34</td>
<td>1.37</td>
<td>5.62</td>
<td>4.02</td>
</tr>
</tbody>
</table>

Effects of Each Uncertain Parameter

<table>
<thead>
<tr>
<th>$V_p$</th>
<th>1.0 $V_{core}$</th>
<th>-1.67</th>
<th>-2.13</th>
<th>-1.65</th>
<th>-1.22</th>
<th>0.06</th>
</tr>
</thead>
<tbody>
<tr>
<td></td>
<td>0.2 $V_{core}$</td>
<td>5.06</td>
<td>4.40</td>
<td>5.02</td>
<td>5.85</td>
<td>0.20</td>
</tr>
<tr>
<td>$Q_p$</td>
<td>0.52%</td>
<td>1.03</td>
<td>-1.86</td>
<td>0.83</td>
<td>5.36</td>
<td>4.18</td>
</tr>
<tr>
<td></td>
<td>0.42%</td>
<td>2.03</td>
<td>-0.78</td>
<td>1.92</td>
<td>5.83</td>
<td>3.62</td>
</tr>
<tr>
<td>$\epsilon_p$</td>
<td>0.8</td>
<td>1.32</td>
<td>-1.42</td>
<td>1.27</td>
<td>4.79</td>
<td>3.14</td>
</tr>
<tr>
<td></td>
<td>0.2</td>
<td>1.10</td>
<td>-1.94</td>
<td>0.76</td>
<td>6.51</td>
<td>5.56</td>
</tr>
</tbody>
</table>
interval of the decay power is narrow in comparison with other parameters.

Buoyancy Driven Recirculation for PWRs

During a high pressure loss of injection sequence, (e.g. station blackout) in a PWR, buoyancy driven recirculation flows may lead to heat transfer from the hot gases leaving the uncovered core to the cooler structures in the reactor vessel and coolant loops. The recirculation flow can manifest itself in several ways; the traditional coolant path through the coolant loops, recirculation within the vessel, recirculation between the upper plenum and the hot side plenum of the steam generator, recirculation through the steam generator (SG) tubes from the hot to cold side plena, and various combinations.

Important consequences of recirculation are creep rupture failure of the RCS loops due to the increased structure temperature. Three locations are of primary concern: the hot leg nozzle because it is exposed to the highest vapor temperature, the pressurizer surge line because it is one-third the hot leg thickness, and the steam generator tubes because they are the thinnest structural boundaries of the RCS. If the steam generator tubes fail, a direct path to the atmosphere may be available through the steam line safety valves allowing fission products to bypass the containment [Bayless, 1988].

Two possibilities should be examined. First, if the loop seal is not cleared, a stratified recirculation flow in the horizontal hot leg is established between the upper plenum of RPV and the inlet plenum of the steam generator due to density differences in the hot leg. The mixing of incoming flows from the hot leg and steam generator tubes occurs in the steam generator inlet plenum. A buoyancy driven flow is established through the steam generator tubes from the inlet to the outlet plenum through some tubes with return flow from the outlet to inlet plenum in others. Figure 3 is a schematic diagram of this process.

Second, if the loop seal is cleared of water due to a pump seal LOCA, the possibility may exist for the gas flow to circulate around the entire RCS, including the cold leg. Then, hot leg recirculation flow does not occur and the SG tubes may fail before the other RCS components. The time when the loop seal is cleared depends strongly on the seal leakage. To make a conclusive statement about this issue, a study of SGTR including proper accounting of the pump seals and when the loop seals are cleared must be performed. The second possibility was not examined in this work.

The time of SGTR is determined using a simplified 1-D model, which was originally described by Wassel et al. (1988), to calculate heatup of the hot leg and steam generator tubes due to recirculation between the upper plenum and the hot side plenum of the steam generator. The purpose of the 1-D analysis is not to resolve this issue, but to describe the important phenomena in this issue using an engineering model which is appropriate for a scoping study. The results of existing analyses by Bayless (1988) and Cha et al. (1989), and experimental results by Stewart et al. (1986) are used to reduce restrictions of the present analysis due to simplifications, which lead to some subjectivity in the SGTR time estimation. Some modifications were made to the Wassel et al [1988] model to evaluate the RCS pipe shell temperatures. The pressurizer surge line and PORV operations are considered in addition to that done by Wassel et al. Many of the assumptions made in the 1-D analysis need to be revisited.

The Larson-Miller parameters for each component were obtained by employing a least-squares fit to the creep rupture data from the INEL results [Harris et al., 1986] for the hot leg
nozzle (A-508 Class 2 carbon steel), hot leg piping (316 stainless steel), and steam generator tubes (Inconel 600). For the surge line, the Larson-Miller parameter for the hot leg piping is used, based on an assumption that the surge line material is same as the hot leg piping material. The temperatures used in the Larson-Miller parameter are the volume average temperature at the hottest locations of the components or the average temperature of the entire component (for the surge line), and the stresses are the hoop stresses in the components. 

Many parameters are used to make the simple model. All of the parameters are essentially uncertain because there are little or no appropriate data. Based on a few preliminary calculations, four influential parameters are selected; the mixing parameter (f), the heat transfer coefficient of the outer wall \((h_{o})\), the heat transfer coefficients of the inner wall \((h_{i})\) and outer wall \((h_{e})\). Unlike previous studies using more detailed models, the uncertain parameter effects are very restricted. Two dominant parameters are selected to make response surfaces for the creep rupture times of each of the components: For the hot leg nozzle and the hot leg pipe, \(h_{e}\) and \(h_{i}\) are selected. For the SG tubes, \(f\) and \(h_{o}\) are selected. For the surge line, the parameters affecting the failure time cannot be investigated because the model for the surge line is too simple. (The model for the surge line has no parameter.) The surge line failure time is always 12575 sec after the accident initiation. For the base case, the selected parameter values were \(f = 0.7, h_{o} = 10W/m^2K\) and \(h_{e} = 10W/m^2K\).

Response surfaces for the rupture time of each component are obtained using the selected uncertain parameters. If the steam generator tubes fail before any other RCS components, including the RPV, SGTR would occur. Values of the calculational points used to generate the response surface are found in Table 3. To obtain a base case, the integration of each of the effects of the uncertain parameters on the rupture time of each component are found using a Monte Carlo method with a sample size of 100,000. Figure 4 shows the rupture time distribution of each component after the initiation of accident. It can be seen that even if the SG tubes do not fail, the other components could fail before vessel failure if the cavity is flooded. When the secondary system is depressurized due to safety relief valve failure, the SGTR time is reduced and probability of the earliest primary system breach being the steam generator tubes is significantly increased. If the mixing in the inlet plenum is poor (i.e. \(f=0.25\)), the SGTR time is advanced and the probability of SGTR may increase significantly. Both the mixing parameter \(f\), and \(h_{o}\) have a considerable affect on the SGTR time distributions. Table 4 presents mean, 5, 50 and 95 percentile values for both intact and failed steam generator safety relief valves.

Table 3. List of Independent Variables and Calculation Points.

<table>
<thead>
<tr>
<th>Uncertain Parameters</th>
<th>Calculation Points</th>
</tr>
</thead>
<tbody>
<tr>
<td></td>
<td>Low</td>
</tr>
<tr>
<td></td>
<td>High</td>
</tr>
<tr>
<td>(h_{o} (W/m^2K))</td>
<td>0.0, 10.0, 20.0</td>
</tr>
<tr>
<td>(h_{i} (% of correlation))</td>
<td>50, 100, 200</td>
</tr>
</tbody>
</table>
III EVALUATION OF A FLOODING STRATEGY FOR A PWR

Deterministic models simulating vessel lower head failure and buoyancy driven recirculation for the evaluation of the lower cavity flooding strategy were used to determine the outcomes of certain events. These two severe accident phenomena are important when evaluating the strategy. The distribution of a particular outcome of the phenomena were determined based on our current understanding of the phenomena. The strategy is then assessed using the influence diagram method with the calculated distributions. The influence diagram established as a part of the UCLA study [Kastenberg et al., 1993] shown in Figure 1, is used with some modification for the assessment.

Discussion of Results from Deterministic Models

Vessel Lower Head Failure

In the influence diagram established in the UCLA work, the probability of vessel breach (VB), $P_{vb}$, is dependent on the success of flooding (WC), whether or not core damage is arrested (CDA) and the RCS pressure (RP) at vessel breach (VB). To determine probability of VB ($P_{vb}$) under a set of conditions such as success of flooding, core damage arrested and any RCS pressure, a 1-D analysis was performed. $P_{vb}$ for the remaining cases can be determined without the above analysis, i.e. the probability is either 1 or 0, or can be obtained from other sources.

According to the 1-D analysis, the most probable lower head failure mechanism is probably creep rupture. When the RCS is at low pressure, the vessel lower head may be stable and the probability of VB is assumed to be 0.0. When the RCS is at high pressure, the lower head may fail, but the time of vessel failure is uncertain. Because of the presence of water outside the lower head, the probability of high pressure melt ejection (HPME) is be low and taken to be 0.0. In this study, the only failure mode is assumed to be bottom head failure. Table 5 shows the conditional probabilities used for the strategy evaluation.

The vessel failure time is affected by parameters such as pool volume, decay heat and corium emissivity, whose values are uncertain. Based on the state of knowledge distribution of these parameters, the distribution of the vessel failure time is obtained. The vessel failure time is used to determine the probability of AC power recovery, P(R), which is represented by node R in the influence diagram. Node R is affected by the presence of water in the cavity (WC); in the UCLA study, node R has no predecessors. If the cavity is flooded (given WC), the value of P(R) is represented by a probabilistic distribution. Otherwise, P(R) is 0.26 as in the UCLA study.

The vessel failure time, $t_v$, obtained by a 1-D analysis is the time at which the corium pool configuration becomes steady. However, the vessel failure time used to determine P(R) is defined as a vessel failure time after the initiation of the accident ($t_{vb}$). It is assumed that 8 hours are required to attain a steady pool configuration in the lower head. Thus, if the cavity is flooded, 8 hours is added to the vessel failure time, i.e. $t_{vb} = t_v + 8$ hours. Otherwise, $t_{vb}$ is 3 hours as the assumed in the UCLA.

If AC power is recovered before $t_{vb}$, the RCS can be depressurized. The load causing creep is removed and the vessel lower head is probably stable. However, because safety
Table 4  Mean Value and 5, 50, and 95 Percentile Values of SGTR Time (in thousands of seconds), under Various Conditions.

<table>
<thead>
<tr>
<th></th>
<th>Mean</th>
<th>5%</th>
<th>50%</th>
<th>95%</th>
<th>Variance</th>
</tr>
</thead>
<tbody>
<tr>
<td>With operating relief valves</td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>Base Case</td>
<td>13.45</td>
<td>11.90</td>
<td>13.24</td>
<td>15.70</td>
<td>1.40</td>
</tr>
<tr>
<td>Effects of Each Uncertain Parameter</td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>f</td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>1.0</td>
<td>15.50</td>
<td>13.60</td>
<td>15.59</td>
<td>17.23</td>
<td>1.11</td>
</tr>
<tr>
<td>0.25</td>
<td>11.93</td>
<td>11.15</td>
<td>11.95</td>
<td>12.59</td>
<td>0.18</td>
</tr>
<tr>
<td>h_{i,a} (W/m²K)</td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>20</td>
<td>14.23</td>
<td>10.61</td>
<td>14.26</td>
<td>17.20</td>
<td>3.39</td>
</tr>
<tr>
<td>0</td>
<td>12.53</td>
<td>12.13</td>
<td>12.39</td>
<td>13.29</td>
<td>0.18</td>
</tr>
<tr>
<td>With failed relief valves (stuck open)</td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>Base Case</td>
<td>12.72</td>
<td>11.91</td>
<td>12.60</td>
<td>13.95</td>
<td>0.42</td>
</tr>
<tr>
<td>Effects of Each Uncertain Parameter</td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>f</td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>1.0</td>
<td>13.83</td>
<td>12.89</td>
<td>13.86</td>
<td>14.72</td>
<td>0.29</td>
</tr>
<tr>
<td>0.25</td>
<td>11.92</td>
<td>11.58</td>
<td>11.92</td>
<td>12.24</td>
<td>0.05</td>
</tr>
<tr>
<td>h_{i,a} (W/m²K)</td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>20</td>
<td>13.13</td>
<td>11.34</td>
<td>13.13</td>
<td>14.70</td>
<td>0.90</td>
</tr>
<tr>
<td>0</td>
<td>12.26</td>
<td>12.01</td>
<td>12.17</td>
<td>12.73</td>
<td>0.07</td>
</tr>
</tbody>
</table>

2.4

Table 5. Conditional Probabilities for node VB in the influence diagram.

<table>
<thead>
<tr>
<th>CDA</th>
<th>RP</th>
<th>WC</th>
<th>P(no VB)</th>
<th>P(HPME)¹</th>
<th>P(BH)²</th>
</tr>
</thead>
<tbody>
<tr>
<td>yes</td>
<td>any</td>
<td>any</td>
<td>1.0</td>
<td>0.0</td>
<td>0.0</td>
</tr>
<tr>
<td>no</td>
<td>high</td>
<td>yes</td>
<td>0.0</td>
<td>0.0</td>
<td>1.0</td>
</tr>
<tr>
<td>no</td>
<td>high</td>
<td>no</td>
<td>0.0</td>
<td>0.79</td>
<td>0.21</td>
</tr>
<tr>
<td>no</td>
<td>low</td>
<td>yes</td>
<td>1.0</td>
<td>0.0</td>
<td>0.0</td>
</tr>
<tr>
<td>no</td>
<td>low</td>
<td>no</td>
<td>0.0</td>
<td>0.0</td>
<td>1.0</td>
</tr>
</tbody>
</table>

Note  
1) HPME = High Melt Ejection (penetration failure)  
2) BH = Bottom Head Failure  
*) The time of vessel failure, t_{VB}, is uncertain except for t_{VB}=3.0 hours.
system availability is in question, further study of the availability of a system for depressurization when AC power recovery occurs should be carried out. The probability, P(R), is optimistic because it is assumed that the system is available and there is no hesitation in operator action.

Table 6. 5, 50, and 95 Percentile Values, Means and Variances of the Probability of AC Power Recovery When the Cavity Is Flooded.

<table>
<thead>
<tr>
<th></th>
<th>95%</th>
<th>50%</th>
<th>5%</th>
<th>mean</th>
<th>variance</th>
</tr>
</thead>
<tbody>
<tr>
<td><strong>Base Case</strong></td>
<td>1.00</td>
<td>0.997</td>
<td>0.782</td>
<td>0.932</td>
<td>8.47E-03</td>
</tr>
<tr>
<td><strong>Effects of Each Uncertain Parameter</strong></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>$V_{p} = V_{core}$</td>
<td>0.783</td>
<td>0.781</td>
<td>0.780</td>
<td>0.781</td>
<td>6.65E-05</td>
</tr>
<tr>
<td>$V_{p} = 0.2V_{core}$</td>
<td>1.00</td>
<td>1.00</td>
<td>1.00</td>
<td>1.00</td>
<td>0.00</td>
</tr>
<tr>
<td>$Q_{p} = 0.52%$</td>
<td>1.00</td>
<td>0.943</td>
<td>0.781</td>
<td>0.909</td>
<td>8.80E-03</td>
</tr>
<tr>
<td>$Q_{p} = 0.42%$</td>
<td>1.00</td>
<td>1.00</td>
<td>0.788</td>
<td>0.951</td>
<td>6.87E-03</td>
</tr>
<tr>
<td>$\varepsilon_{p} = 0.8$</td>
<td>1.00</td>
<td>0.993</td>
<td>0.782</td>
<td>0.929</td>
<td>8.67E-03</td>
</tr>
<tr>
<td>$\varepsilon_{p} = 0.8$</td>
<td>1.00</td>
<td>0.932</td>
<td>0.780</td>
<td>0.905</td>
<td>9.23E-03</td>
</tr>
</tbody>
</table>

For modeling the recovery of AC power, only off-site power is modeled. It is assumed that diesel generators can not be recovered easily after their failure. Iman et al. (1988) gives a Weibull distribution for non-recovery time of off-site AC power;

$$P[T > t] = \exp[-(\lambda t)^{\beta}]$$

where $\lambda = 0.44$ and $\beta = 0.72$. The timing of core uncover and core slump are 2.15 and 3.0 hours after TMLB' initiation, respectively, which are the same times as those in the UCLA study.

Table 6 shows the mean values, 5, 50 and 95 percentile values and variances of P(R) distributions under various conditions when the cavity is flooded. As expected, the pool volume ($V_p$) is the parameter most influencing on the distribution of P(R). The other parameters affect the distribution little.

**RCS Boundary Failure due to the Recirculation**

The distribution of the time to SGTR due to buoyancy driven recirculation was obtained, using state-of-knowledge distributions for two parameters; mixing parameter, $f$, and heat transfer coefficient of SG tube outer wall, $h_{out}$, whose values are uncertain. Two cases are considered here; 1) closed safety relief valves, and 2) open safety relief valves.
Closed safety relief valves

According to the 1-D analysis results, the steam generator tubes may fail at least 4 hours after the initiation of accident while the vessel failure time may be 3 hours after the initiation of accident when the lower head is not cooled by water. When the cavity is not flooded, neither SGTR nor HSF (hot leg or surge line failure) may occur because vessel failure before any upper RCS components fail is probable. When the lower head is cooled by water, the time of SGTR due to the recirculation may be earlier than the time to reach a steady corium pool in the vessel lower head. Thus, the 1-D analysis results represent the phenomena related to node ESGTR in the influence diagram. The node ESGTR affects HSF, RP and SGTR nodes.

In the analysis, for the surge line, parameters affecting the time of surge line failure could not be investigated because the model for the surge line is too simple. As a result, the surge line failure time is always 12575 sec after the initiation of accident. To consider the uncertainty in the hot leg or surge line failure time (t_{HFS}), a state-of-knowledge distribution of t_{HFS} is specified, based on this analysis.

The state-of-knowledge distribution of t_{HFS} and its basis are as follows: The calculated surge line failure time is conservative. The probability of HSF at the time of 12575 sec is assumed to be 0.5. After 12700 sec, the hot leg nozzle would fail and the probability of HSF becomes high. The lower and upper bounds are arbitrarily assumed to be 12400 sec and 13000 sec, respectively. A 1/2 chance is assigned to the 12400-12575 sec interval and a 0.3 chance is assigned to 12575-12700 sec. A 0.2 chance is assigned to 12700-13000 sec.

By using the calculated SGTR time and the specified distribution of t_{HFS}, the probability of ESGTR (P_{ESGTR}) can be determined. At a calculated SGTR time, P_{ESGTR} is the probability of no HSF. Because either HSF or ESGTR may occur under any conditions, the conditional probability of HSF, given no ESGTR is assumed to be 1.0 while given ESGTR, it is assumed to be 0.0. To describe the phenomena found in the 1-D analysis, two arcs should be added to the influence diagram; from node WC to nodes ESGTR and HSF, i.e. WC affects ESGTR and HSF. Figure 1 shows the modified influence diagram. The conditional probabilities of ESGTR and HSF are represented in Table 7.

<table>
<thead>
<tr>
<th>WC</th>
<th>ESGTR</th>
<th>HSF</th>
</tr>
</thead>
<tbody>
<tr>
<td>yes</td>
<td>P_{ESGTR}</td>
<td>0.0</td>
</tr>
<tr>
<td>no</td>
<td>0.0</td>
<td></td>
</tr>
</tbody>
</table>

Table 7: Conditional Probabilities for nodes ESGTR and HSF in the modified diagram.

Note *) The distribution of P_{ESGTR} represents the parameter uncertainty in the 1-D analysis.

Table 8 shows 5, 50 and 95 percentile values, means and variances of P_{ESGTR}
distributions under various conditions. Because the interval of the assumed distribution of \( t_{\text{HSP}} \)

is narrow, most values of \( P_{\text{ESGTR}} \) are 0 or 1, i.e. most calculated SGTR times are either longer

than the higher bound of the \( t_{\text{HSP}} \) distribution or shorter than its lower bound. As expected,

both \( f \) and \( h_{\text{c}} \) have a strong influence on the distribution of \( P_{\text{ESGTR}} \).

**Open safety relief valves**

Due to the safety relief valve failure, the secondary system may be depressurized
during the accident. When the secondary system is depressurized, the pressure difference
between inside and outside steam generator tubes increases and the time to SGTR is advanced.

**Table 8** 5, 50, and 95 Percentile Values, Means and Variances of the

Probability of ESGTR When the Cavity Is Flooded

<table>
<thead>
<tr>
<th></th>
<th>95%</th>
<th>50%</th>
<th>5%</th>
<th>mean</th>
<th>variance</th>
</tr>
</thead>
<tbody>
<tr>
<td>With operating relief valves</td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>Base Case</td>
<td>1.00</td>
<td>0.00</td>
<td>0.00</td>
<td>0.250</td>
<td>0.161</td>
</tr>
<tr>
<td>Effects of Each Uncertain Parameter</td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>( f=1 )</td>
<td>0.00</td>
<td>0.00</td>
<td>0.00</td>
<td>0.000</td>
<td>0.000</td>
</tr>
<tr>
<td>( f=.25 )</td>
<td>1.00</td>
<td>1.00</td>
<td>0.478</td>
<td>0.940</td>
<td>0.026</td>
</tr>
<tr>
<td>( h_{\text{c}}=20 )</td>
<td>1.00</td>
<td>0.00</td>
<td>0.00</td>
<td>0.174</td>
<td>0.140</td>
</tr>
<tr>
<td>( h_{\text{c}}=10 )</td>
<td>1.00</td>
<td>1.00</td>
<td>0.00</td>
<td>0.655</td>
<td>0.182</td>
</tr>
<tr>
<td>(W/m²K)</td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>With failed relief valves</td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>Base Case</td>
<td>1.00</td>
<td>0.44</td>
<td>0.00</td>
<td>0.496</td>
<td>0.194</td>
</tr>
</tbody>
</table>

The effect of secondary system depressurization is only on the time to SGTR. As expected,

\( P_{\text{ESGTR}} \) is greater than that when the secondary system is not depressurized. Both \( f \) and \( h_{\text{c}} \)

have strong influence on the distribution of \( P_{\text{ESGTR}} \).

**Evaluation of the Flooding Strategy for PWR**

The flooding strategy modeled by the influence diagram described in Figure 1, is

evaluated. In order to evaluate an accident management strategy, a suitable criterion is needed
against which to compare the alternatives. This criterion is associated with the value node of
the influence diagram, and for this strategy there are several possibilities: early fatalities, the
conditional probability of early containment failure, economic loss, etc. The criterion should
be chosen in such a way that all factors of the assessment are captured, including adverse
effects and secondary mitigative effects. In this study, risk (early and late fatalities) is used as
a criterion, as was done in the UCLA study.

A best estimate assessment of the flooding strategy is performed, as done in the UCLA
study. The best estimate values for the distributed values (\( P(R), P_{\text{ESGTR}} \)) are the expected values
(mean values) of the uncertainty distributions of the base cases, which are given in Tables 6 and 8. The updated conditional probabilities for the evaluation of the modified influence diagram are given in Tables 5 and 7. The remaining probabilities needed for the evaluation are those used in the UCLA study.

The result is shown in Table 9. The early fatalities are reduced when the cavity is flooded because when the decision is 'Flood', the marginal probability of vessel failure is significantly reduced even though $P_{BS0TR}$ increases. (The marginal probability of early containment failure (ECF) decreases as the marginal vessel failure probability decreases and the early fatalities due to ECF is much larger than the early fatalities due to SGTR.) The preferred decision is 'Flood' when the measure of the off-site consequences is early fatalities. However, the late cancer fatalities due to ECF is comparable with the late fatalities due to SGTR. Late cancer fatalities increase when the cavity is flooded. The preferred decision is 'Do Nothing' when the measure of the off-site consequences is late fatalities. The 'Certainty' in Table 9 is defined as how much it is certain that 'Flood' results in a lower early (or late) fatalities than 'Do Nothing', given the state-of-knowledge distributions of the uncertain parameters and the deterministic models.

The estimated probability distributions represent the state-of-knowledge uncertainty in the parameters used in the deterministic models. The uncertainty distributions of the uncertain parameters are propagated through the influence diagram, and the result is a distribution over the values of the risks (early and late fatalities). There are several methods developed for the propagation of uncertainty; the method employed here is Monte Carlo simulation with a sample size of 100,000. The uncertainty in early and late fatalities when 'Do Nothing' is a decision is not investigated in this study.

The results of a base case calculation using all assigned distributions of the uncertain parameters are shown in Tables 9 and 10. It can be seen that the preferred decision depends on the values of uncertain parameters. One can make a statement that it is 78% certain that 'Flood' results in lower early (or 67% for late) fatalities than 'Do Nothing', given the state-of-knowledge distributions of the uncertain parameters and the deterministic models. When the

<table>
<thead>
<tr>
<th>Early Fatalities</th>
<th>Do Nothing</th>
<th>Flooding$^1$</th>
<th>Flooding$^2$</th>
<th>Certainty</th>
</tr>
</thead>
<tbody>
<tr>
<td></td>
<td>3.45E-2</td>
<td>2.41E-2</td>
<td>2.48E-2</td>
<td>100%</td>
</tr>
<tr>
<td>Late Fatalities</td>
<td>4.99E+1</td>
<td>2.33E+2</td>
<td>4.47E+2</td>
<td>0%</td>
</tr>
</tbody>
</table>

Note 1) When the secondary system pressure remains high
2) When the secondary system pressure remains low

relief valves are open, as expected, the certainty is reduced.

To investigate the effects of each uncertain parameter used in the deterministic models on risk, the uncertainty in each uncertain parameter is reduced to zero, one at a time, and the remaining distributions are propagated through the influence diagram. For each parameter being studied, a high or low value (+1 or -1 in normalized values) is used. Figures 10, 11,
and 12 show the risk distributions under various conditions of the uncertain parameters in the models for vessel failure and SGTR phenomena. The uncertain parameters in the model for SGTR (mixing parameter, \( f \), and heat transfer coefficient of steam generator tube outer wall, \( h_{w}\)) affect the risk significantly while there is no significant effect of uncertain parameters in the model for vessel failure.

Tables 10 and 11 show the 5, 50 and 95 percentile values, means and variances of early and late fatalities, respectively. As shown in Figure 10, even though the pool volume is the most influential parameter on the vessel failure time, it does not considerably affect the risk distributions because the probable interval of \( P(R) \), 0.78-1.0, is narrow. Two parameters in the model for SGTR, \( f \) and \( h_{w}\), are influential on the risk, regardless of the secondary system pressure. When the value of \( f \) is fixed, the most reduction in variance is seen. It can be concluded that the mixing parameter, \( f \), is the most important parameter in terms of contributing to the uncertainty in the risk. If one believes the models used in the present study and the assumed distributions for uncertain parameters, one can conclude that buoyancy driven recirculation and the resulting upper RCS component failure are the most urgent study subjects if a more reliable assessment of the result of the flooding strategy is desired.

IV. CONCLUSIONS

Estimates of the effectiveness of the flooding strategy were improved by better treatment of selected phenomena. According to the 1-D conservative analysis of vessel failure, with a flooded cavity, only creep rupture may be a possible vessel lower head failure mechanism. When the RCS is at low pressure, the vessel lower head may be stable and the probability of VB is probably 0.0. When the RCS is at high pressure, the lower head may fail, but the time of vessel failure is uncertain, which determines the probability of AC power recovery before vessel failure.

According to the 1-D analysis for upper RCS component failures due to the buoyancy driven recirculation, the steam generator tubes may fail about 4 hours after the initiation of the accident while the vessel failure time may be 3 hours after the initiation of the accident when the lower head is not cooled by water. When the cavity is not flooded, neither SGTR nor HSF (hot leg or surge line failure) may occur because vessel failure before any upper RCS components fail is probable, regardless of the secondary system pressure. When the lower head is cooled by water, the time of SGTR due to the recirculation may be earlier than the time of attainment for a steady corium pool in the vessel lower head. Thus, the flooding strategy is not always effective.

The effect of the secondary system pressure on the risk results is not negligible. When the secondary system is depressurized, the time to SGTR is predicted to be reduced and the risk results are predicted to be increased. Further study about the status of the secondary system pressure is needed.

When the cavity is flooded, the RCS pressure is probably reduced before vessel failure due to AC power recovery or the upper RCS failure. When AC power is recovered before vessel failure, the RCS can be depressurized if the system is available. Even though AC power is not recovered before vessel failure, the RCS may be at low pressure due to the upper RCS component failure because the time of the upper RCS component failure may be much earlier.
### Table 10  Distributions of Early Fatalities under Various Conditions.

<table>
<thead>
<tr>
<th></th>
<th>95%</th>
<th>50%</th>
<th>5%</th>
<th>mean</th>
<th>variance</th>
<th>Certainty</th>
</tr>
</thead>
<tbody>
<tr>
<td><strong>With operating relief valves</strong></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>Base Case</td>
<td>5.40E-2</td>
<td>1.42E-2</td>
<td>1.42E-2</td>
<td>2.35E-2</td>
<td>2.47E-4</td>
<td>77.0%</td>
</tr>
</tbody>
</table>

**Effects of Each Uncertain Parameter**

**in Model for Vessel Failure**

<table>
<thead>
<tr>
<th>$V_p$</th>
<th>95%</th>
<th>50%</th>
<th>5%</th>
<th>mean</th>
<th>variance</th>
<th>Certainty</th>
</tr>
</thead>
<tbody>
<tr>
<td>$1.0V_{core}$</td>
<td>5.46E-2</td>
<td>1.42E-2</td>
<td>1.42E-2</td>
<td>2.37E-2</td>
<td>2.55E-4</td>
<td>76.9%</td>
</tr>
<tr>
<td>$0.2V_{core}$</td>
<td>5.37E-2</td>
<td>1.42E-2</td>
<td>1.42E-2</td>
<td>2.36E-2</td>
<td>2.47E-4</td>
<td>77.0%</td>
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</tbody>
</table>

**in Model for Upper RCS Failure due to Buoyancy Driven Recirculation**

<table>
<thead>
<tr>
<th>$f$</th>
<th>95%</th>
<th>50%</th>
<th>5%</th>
<th>mean</th>
<th>variance</th>
<th>Certainty</th>
</tr>
</thead>
<tbody>
<tr>
<td>1.0</td>
<td>1.42E-2</td>
<td>1.42E-2</td>
<td>1.42E-2</td>
<td>1.42E-2</td>
<td>0.00</td>
<td>100%</td>
</tr>
<tr>
<td>0.25</td>
<td>5.46E-2</td>
<td>5.37E-2</td>
<td>3.35E-2</td>
<td>5.16E-2</td>
<td>4.04E-5</td>
<td>5.9%</td>
</tr>
<tr>
<td>$h_{oe}[W/m^2K]$</td>
<td>95%</td>
<td>50%</td>
<td>5%</td>
<td>mean</td>
<td>variance</td>
<td>Certainty</td>
</tr>
<tr>
<td>20</td>
<td>5.40E-2</td>
<td>1.42E-2</td>
<td>1.42E-2</td>
<td>2.11E-2</td>
<td>2.23E-4</td>
<td>82.6%</td>
</tr>
<tr>
<td>0</td>
<td>5.46E-2</td>
<td>5.37E-2</td>
<td>1.42E-2</td>
<td>4.06E-2</td>
<td>2.87E-4</td>
<td>34.2%</td>
</tr>
</tbody>
</table>

**With failed relief valves**

<table>
<thead>
<tr>
<th></th>
<th>95%</th>
<th>50%</th>
<th>5%</th>
<th>mean</th>
<th>variance</th>
<th>Certainty</th>
</tr>
</thead>
<tbody>
<tr>
<td>Base Case</td>
<td>5.45E-2</td>
<td>3.16E-2</td>
<td>1.47E-2</td>
<td>3.38E-2</td>
<td>3.09E-4</td>
<td>51.4%</td>
</tr>
<tr>
<td>Table 11</td>
<td>Distributions of Late Fatalities under Various Conditions.</td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>---------</td>
<td>-------------------------------------------------------------</td>
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<td></td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td></td>
<td>95%</td>
<td>50%</td>
<td>5%</td>
<td>mean</td>
<td>variance</td>
<td>Certainty</td>
</tr>
<tr>
<td>With operating relief valves</td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>Base Case</td>
<td>8.73E+2</td>
<td>2.05E+1</td>
<td>2.05E+1</td>
<td>2.21E+2</td>
<td>1.13E+5</td>
<td>66.2%</td>
</tr>
<tr>
<td>Effects of Each Uncertain Parameter</td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>in Model for Vessel Failure</td>
<td></td>
<td></td>
<td></td>
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<td></td>
<td></td>
</tr>
<tr>
<td>$V_r$</td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>1.0V_{core}</td>
<td>8.74E+2</td>
<td>2.05E+1</td>
<td>2.05E+1</td>
<td>2.21E+2</td>
<td>1.14E+5</td>
<td>66.3%</td>
</tr>
<tr>
<td>0.2V_{core}</td>
<td>8.72E+2</td>
<td>2.05E+1</td>
<td>2.05E+1</td>
<td>2.24E+2</td>
<td>1.15E+5</td>
<td>66.3%</td>
</tr>
<tr>
<td>in Model for Upper RCS Failure due to Buoyancy Driven Recirculation</td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>$f$</td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>1.0</td>
<td>2.05E+1</td>
<td>2.05E+1</td>
<td>2.05E+1</td>
<td>2.05E+1</td>
<td>0.00</td>
<td>100%</td>
</tr>
<tr>
<td>0.25</td>
<td>8.74E+2</td>
<td>8.72E+2</td>
<td>4.35E+2</td>
<td>8.22E+2</td>
<td>1.87E+4</td>
<td>0%</td>
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<tr>
<td>$h_a$ (W/m²K)</td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>20</td>
<td>8.73E+2</td>
<td>2.05E+1</td>
<td>2.05E+1</td>
<td>1.70E+2</td>
<td>1.02E+5</td>
<td>81.2%</td>
</tr>
<tr>
<td>0</td>
<td>8.74E+2</td>
<td>8.72E+2</td>
<td>2.05E+1</td>
<td>5.86E+2</td>
<td>1.32E+5</td>
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<tr>
<td>With failed relief valves</td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>Base Case</td>
<td>8.74E+2</td>
<td>3.95E+2</td>
<td>2.05E+1</td>
<td>4.41E+2</td>
<td>1.42E+5</td>
<td>31.9%</td>
</tr>
</tbody>
</table>

than the time to vessel failure. Thus, the vessel failure is not probable when the reactor cavity is flooded. It can be concluded that the vessel lower head would be stable when the cavity is flooded but the effectiveness of the flooding strategy is in question due to phenomenological uncertainty.

**REFERENCES**


Siu, N. and Apostolakis, G., On the Quantification of Modeling Uncertainties, 8th


Figure 1. Modified influence diagram for reactor cavity flooding strategy applied to Surry TLMB' sequences.
Figure 2. Schematic diagram of a relocated core
Figure 3. Schematic diagram of buoyancy driven recirculation between RPV upper plenum and steam generator
Figure 4. Distributions of creep rupture times for three Components
ANALYSIS OF SEVERE ACCIDENT PHENOMENOLOGY AND NEED FOR
COMPUTERIZED ACCIDENT MANAGEMENT SUPPORT

by

Ø. Berg, A. Sørensen, IFE
I. Lindholm, J. Miettinen, E. Peikarinen, VTT Energy
H. Sjövall, S. Koski, TVO

INTRODUCTION

The objective of the work presented in this paper is to enhance the understanding of selected phenomena related to the progression of a hypothetical severe accident in Nordic reactors. The knowledge gained in the studies will help the power utilities and the regulatory authorities in planning and execution of severe accident management. Further, the objective is to study the information needs in accident scenarios and to investigate how to design adequate computerized tools and information systems for assistance in accident management.

USE OF CAMS DURING ACCIDENTS

- Use before Core Degradation

The accident management for pressurized water reactors, beyond design basis initiating event, requires operator actions on primary or secondary side in order to get cooling water into the primary circuit to prevent core heatup. Operator actions are required in the form of primary feed, primary bleed, secondary feed or secondary bleed. The best accident management actions could be estimated on the basis of the plant status, for which the key parameters are the primary pressure, water inventory in the pressure vessel, secondary pressure and water inventory in the steam generators. Based on extensive studies for different thermohydraulic scenarios a decision making procedure could be developed for the pre-severe accident, where different operator actions are proposed and by predictive simulations the consequences of selected actions are studied. This kind of simulation is possible by using a thermohydraulic model with a calculation accuracy of typical safety analysis programs. The thermohydraulic model of APROS is a sufficient tool for the purpose and its capability has been demonstrated in the CAMS project, see ref. 2, where the development has focused on the prediction and tracking before core damage take place.

The accident management for boiling water reactors prior to onset of core damages includes similar actions for injecting coolant into the vessel. Available operator actions are the attempts for injecting water into the vessel and reduction of the pressure in the RCS for activation of low pressure injection. The accident management procedures for ASEA-type BWR with internal circulation pumps are considered in the following.

After the initiation of an accident the main task of the operators is according to the symptom based Emergency Operating Procedures (EOPs) to ensure that the reactor is sufficiently cooled. Identifying the cause of the accident will come later. CAMS could support the operators with information in the first phase of an accident. Plant instrumentation gives initial data on temperatures, pressures and water levels in the primary system, containment and auxiliary building to be fed as input to CAMS. Based on that information and using an expert system and thermal hydraulic models implemented in CAMS the cause of the accident may be rapidly identified and the further progression of the accident predicted and the required correct operator actions performed. Considering the rather mature state of the thermal hydraulic
modelling, the results are expected to be reliable. CAMS should, however, be used only as an information tool for operators. No automatic accident management action should be based on the results.

For mitigation of the consequences of a severe accident a methodology combining a process diagnostics based on a few measured parameters and predictive simulation could be proposed similarly with the methodology for accident prevention. The main objective of this paper is to evaluate how this kind of operator support tool for severe accident mitigation could be realized if reliable models for severe accident analysis were available.

- Use during Severe Accident

If an accident develops into the severe accident area, CAMS should be designed to include models which can handle core recovery, predict core heat-up, start of core degradation and relocation and risk for vessel failure (even if the calculations of the timing and mode of the vessel failure are very uncertain at this stage). This should provide more information on the accident progression. The severe accident information is best used by the technical support centre which has competence to deal with the considerable uncertainties in the results.

ANALYSIS OF SEVERE ACCIDENT PHENOMENOLOGY

The present example studies focus on evaluation of consequences of reflooding of an overheated, partly damaged core, see ref. 1. The reference plant for the studies is TVO I/II owned and operated by Teollisuuden Voima Oy (TVO) and located on the west coast of Finland. The two units of the TVO NPP each has 2160 MWth boiling water reactors designed by ABB Atom. (Net electric output of 710 MWe). The containment is of the pressure suppression type and is filled with nitrogen during operation.

The base accident scenario analysed is initial loss of power with or without successful depressurisation of the reactor coolant system (RCS). Station blackout as initiating event is a major contributor to the core degradation frequency according to PSA level 1 for TVO. Timing for the recovery of the external power supply and start of core reflooding is varied by using different maximum cladding temperatures, 1400 K, 1600 K, 1800 K and 2000 K, as criterion. In the case of power restoration after a long blackout, the auxiliary feedwater system is the only emergency cooling system to start automatically with the capability to inject into high counterpressure. In all analysed cases the auxiliary feedwater system was assumed to operate with half capacity injecting water to the downcomer.

The studies were performed using three different applicable severe accident computer codes: MAAP4, MELCOR 1.8.3 and SCDAP/RELAP5/Mod 3.1.

In the analysed high-pressure reflooding scenarios all codes predicted a fully quenched end state of the core. In the situation, with the earliest power recovery time (1400 K), the upper part of the core was oxidised, but core deformations were prevented. In other high pressure cases, minor material melting took place in the top third of the core.

In the cases with successful depressurisation of the RCS the core damages were much larger than in the respective high-pressure cases. The low-pressure cases resulted in major core damage and formation of an unquenched rubble bed or melt pool supported by crust in the lower half of the core. The failure of the supporting crust will result in further relocations of material. This model follows the logic obtained from the TMI-2 accident.
Fig. 1a. End state of TVO I/II core in low pressure TB case with reflooding at max. clad temperature of 1600 K. MELCOR prediction.

Fig. 1b. End state of TVO I/II core in low pressure TB case with reflooding at max. clad temperature of 1600 K. MAAP4 prediction.

Fig. 2a. End state of TVO I/II core in high pressure TB case with reflooding at max. clad temperature of 1600 K. MELCOR prediction.

Fig. 2b. End state of TVO I/II core in high pressure TB case with reflooding at max. clad temperature of 1600 K. MAAP4 prediction.
Figures 1 and 2 show schematically the end state of the core in high- and low-pressure scenarios respectively, where reflooding was started at 1600 K.

Table 1 shows the key results from the low-pressure cases and Table 2 from the high-pressure cases.

### Table 1. Key results from the low pressure reflooding scenarios for TVO I/II.

<table>
<thead>
<tr>
<th>Case</th>
<th>Code</th>
<th>No reflooding</th>
<th>1400 K</th>
<th>1600 K</th>
<th>1800 K</th>
<th>2000 K</th>
</tr>
</thead>
<tbody>
<tr>
<td>Start of reflooding</td>
<td>MELCOR</td>
<td>-</td>
<td>5631 s</td>
<td>5950 s</td>
<td>6443 s</td>
<td>6959 s</td>
</tr>
<tr>
<td></td>
<td>MAAP4</td>
<td>-</td>
<td>4800 s</td>
<td>5220 s</td>
<td>5400 s</td>
<td>5810 s</td>
</tr>
<tr>
<td></td>
<td>SCDAP/R5</td>
<td>-</td>
<td>4825 s</td>
<td>4900 s</td>
<td>5380 s</td>
<td>-</td>
</tr>
<tr>
<td>In-vessel H₂ production</td>
<td>MELCOR</td>
<td>424 kg</td>
<td>373 kg</td>
<td>445 kg</td>
<td>413 kg</td>
<td>500 kg</td>
</tr>
<tr>
<td></td>
<td>MAAP4</td>
<td>110 kg</td>
<td>240 kg</td>
<td>450 kg</td>
<td>410 kg</td>
<td>490 kg</td>
</tr>
<tr>
<td></td>
<td>SCDAP/R5</td>
<td>100* kg</td>
<td>162* kg</td>
<td>188* kg</td>
<td>82* kg</td>
<td>-</td>
</tr>
<tr>
<td>Core plate failure</td>
<td>MELCOR</td>
<td>756 s</td>
<td>-</td>
<td>-</td>
<td>-</td>
<td>7506 s</td>
</tr>
<tr>
<td></td>
<td>MAAP4</td>
<td>8600 s</td>
<td>-</td>
<td>-</td>
<td>-</td>
<td>-</td>
</tr>
<tr>
<td></td>
<td>SCDAP/R5</td>
<td>N/A</td>
<td>517 s</td>
<td>N/A</td>
<td>N/A</td>
<td>-</td>
</tr>
<tr>
<td>RPV failure</td>
<td>MELCOR</td>
<td>7658 s</td>
<td>-</td>
<td>-</td>
<td>-</td>
<td>7763 s</td>
</tr>
<tr>
<td></td>
<td>MAAP4</td>
<td>8800 s</td>
<td>-</td>
<td>-</td>
<td>-</td>
<td>-</td>
</tr>
<tr>
<td></td>
<td>SCDAP/R5</td>
<td>N/A</td>
<td>N/A</td>
<td>N/A</td>
<td>N/A</td>
<td>N/A</td>
</tr>
</tbody>
</table>

* still continuing at the end of calculation

### Table 2. Key results from the high pressure reflooding scenarios for TVO I/II.

<table>
<thead>
<tr>
<th>Case</th>
<th>Code</th>
<th>No reflooding</th>
<th>1400 K</th>
<th>1600 K</th>
<th>1800 K</th>
<th>2000 K</th>
</tr>
</thead>
<tbody>
<tr>
<td>Start of reflooding</td>
<td>MELCOR</td>
<td>-</td>
<td>4268 s</td>
<td>4504 s</td>
<td>4590 s</td>
<td>4600 s</td>
</tr>
<tr>
<td></td>
<td>MAAP4</td>
<td>-</td>
<td>4250 s</td>
<td>4390 s</td>
<td>4420 s</td>
<td>4440 s</td>
</tr>
<tr>
<td></td>
<td>SCDAP/R5</td>
<td>-</td>
<td>4210 s</td>
<td>4500 s</td>
<td>4685 s</td>
<td>4737 s</td>
</tr>
<tr>
<td>In-vessel H₂ production</td>
<td>MELCOR</td>
<td>730 kg</td>
<td>32 kg</td>
<td>90 kg</td>
<td>90 kg</td>
<td>90 kg</td>
</tr>
<tr>
<td></td>
<td>MAAP4</td>
<td>500 kg</td>
<td>63 kg</td>
<td>150 kg</td>
<td>140 kg</td>
<td>140 kg</td>
</tr>
<tr>
<td></td>
<td>SCDAP/R5</td>
<td>333 kg</td>
<td>11 kg</td>
<td>27 kg</td>
<td>77 kg</td>
<td>100 kg</td>
</tr>
<tr>
<td>Core plate failure</td>
<td>MELCOR</td>
<td>10409 s</td>
<td>-</td>
<td>-</td>
<td>-</td>
<td>-</td>
</tr>
<tr>
<td></td>
<td>MAAP4</td>
<td>9980 s</td>
<td>-</td>
<td>-</td>
<td>-</td>
<td>-</td>
</tr>
<tr>
<td></td>
<td>SCDAP/R5</td>
<td>N/A</td>
<td>-</td>
<td>-</td>
<td>-</td>
<td>-</td>
</tr>
<tr>
<td>RPV failure</td>
<td>MELCOR</td>
<td>10503 s</td>
<td>-</td>
<td>-</td>
<td>-</td>
<td>-</td>
</tr>
<tr>
<td></td>
<td>MAAP4</td>
<td>9980 s</td>
<td>-</td>
<td>-</td>
<td>-</td>
<td>-</td>
</tr>
<tr>
<td></td>
<td>SCDAP/R5</td>
<td>N/A</td>
<td>-</td>
<td>-</td>
<td>-</td>
<td>-</td>
</tr>
</tbody>
</table>
Comparing the results of the calculated high and low pressure cases, one observes that the depressurization of RCS leads to earlier and more severe core damage than the base case without depressurization. This suggests that the RCS should not be depressurized too early. On the other hand, delay in the opening of the ADS valves may, once the major core melt and slumping starts, eventually lead to high pressure melt ejection. The ADS valves fail to operate at a later stage of the accident or the supposed reflooding of the core is not successful in the assumed time frame. To avoid the failure of RPV at high pressure the depressurization should be performed in due time. The calculations showed that the peak cladding temperature (Fig. 3) was still below the design basis criteria of 1200 °C at 1 h, when the ADS valves were assumed to be opened.

The core reflooding studies presented in this paper showed that there are considerable uncertainties in the results. A slight temperature change at the start of the reflooding could have a significant impact on accident progression. MAAP4, MELCOR and SCDAP/RELAP5 predicted the progression of the core damage in different ways. MELCOR predicted the fragmentation of the fuel and formation of a rubble bed with no fuel or cladding slumping into the lower plenum in the majority of the cases. Some of the code sensitivity parameters are anticipated to have effect on the results. MAAP4 in turn favours formation of a melt pool in the core, with no material relocation to the lower plenum, when core is reflooded. Also MAAP4 results are sensitive to user-input parameters. A SCDAP/RELAP5 calculation resulted in melt pool formation and rapid slumping of material into the lower head. The material relocation to the lower head was, however, controlled by user specified input parameters in SCDAP/RELAP5.

The calculational results presented in this report, even if they show some similar trends, may partly be due to inadequate knowledge about the nature of the phenomena concerned. In general when discussing about reflooding of a severely damaged core, further information about the pertinent physical phenomena occurring during the quenching process is needed in order to evaluate the adequacy of the present calculational models for applications in the context of accident management. As a conclusion it can be said that the severe accident codes contain large uncertainties regarding severe accident phenomena. Calculational results are sensitive to user-input parameters.
COMPUTERIZED ACCIDENT MANAGEMENT SUPPORT

Before being capable to perform predictive simulation of a severe accident, the computerized system should diagnose the existing plant status with the help of the available instrumentation. For the cases studied here, the electric power used by the circulation pumps and feedwater pumps was completely lost. High-pressure injection, serving as an auxiliary feedwater injection, too, is assumed to start after the recovery of power. All these failure assumptions are possible to be identified by the plant instrumentation.

Some severe accident scenarios could be initiated by a feedwater line rupture or steam line rupture. In these cases the leak-size prediction requires special calculation tools. If the boundary conditions in a severe accident situation can be identified, the tracking or predictive simulation could be possible, if reliable physical models are available for the purpose. The phenomenological requirements for the simulation code could be based on experience gained with several severe accident analysis codes.

The analyses with the applied different computer codes suggest that the key parameters describing the process status during a severe accident are the system pressure, the liquid inventory in the vessel, the maximum core temperature, the degree of core degradation, the status of core support plate and the status of the pressure vessel bottom head. The degree of core degradation could be roughly estimated from the amount of hydrogen generated in the core. The temperature of the core support plate could be used for estimation, when molten material relocates to the lower head. If the accident status is known in the form of key parameters, a decision making procedure can be proposed, where the applicable operator actions are the vessel pressure control, water injection into the vessel and pedestal flooding in the containment.

However, the prediction should be reliable enough and the comparisons by using different analysis codes indicate that at present no code alone can predict reliably all possible phases of the severe accident progression. The long development history of MAAP, MELCOR and SCDAP/RELAP5 has demonstrated the complexity of the phenomenology in severe accidents. That is why a special methodology for operator support could be proposed, where for process tracking and prediction different parallel analysis tools would be available. The operator could select e.g. the worst prediction, average prediction or the combination of the two most probable results. The decision-making criteria might be a variable which describes the estimated state.

In the accident situation a proper, reliable monitoring system would be useful for support to the operator, for example visualising the core temperatures and coolant level in the vessel. As a simulation tool it is applicable in training of accident management in severe accident situations. In TVO I/II case where the operator manually initiates the opening of the ADS valves, a good monitoring system would be beneficial, complementing the emergency operation procedures. Another operator-initiated event at TVO I/II is the flooding of the lower drywell with the suppression-pool water. Pedestal flooding is a precaution for possible RPV melt-through to guarantee core melt coolability in the containment. On the other hand, pedestal flooding reduces the heat-sink capacity of the suppression pool, particularly in a station-blackout scenario, where the suppression-pool heat-removal system is not operating.

In 1992 the OECD Halden Reactor Project started a project on Computerised Accident Management Support (CAMS). Through prototype development and human factors experiments one provides knowledge, requirements and experience concerning important
information needs during accident scenarios and how to design and implement computer-based systems supporting accident management. Support is offered in identification of plant state, in assessment of the future development of the accident, and in planning of accident mitigation strategies.

In an accident situation, the user would be reluctant to make use of a system with which he is not familiar. CAMS should also offer support in normal situations, and its accident-handling capabilities should come as a natural extension. The current version of CAMS emphasises support in the initial phase of an accident where the operator still can influence the course of the accident.

The research focuses on application of on-line simulation, strategy generation, and user-interface design. Although simulators have been applied for a long time as off-line tools in accident analysis, the utilisation of simulators and mathematical models for on-line estimation and prediction of process behaviour is limited. In particular, the use of tracking simulation needs further investigation before robust and practical applications can be made.

However, the great potential of on-line simulation technology is obvious in accident situations, where information from process instruments deteriorates and prognosis of accident progression is difficult to make based on qualitative information. Instrumentation data is, however, necessary to verify the predictions.

The user interface of CAMS is developed with the objective to improve the operator's understanding of process behaviour during accident situations. Process-oriented pictures which show energy flow and mass flow and balances in the primary system and containment have been developed. The plant overview and key trend parameters from the reactor and the containment are shown in the three pictures (Figures 4, 5 and 6) taken from the CAMS prototype.

A first prototype of CAMS covers the predictive mode of operation and comprises a predictive simulator, a strategy generator, an interactive graphical user interface and a data communication system supporting a distributed object-oriented data base management.

The predictive simulator has been developed by means of the simulator-development tool APROS from VTT/IVO, Finland. It comprises the main process parts of the BWR at Forsmark Unit 2, Sweden. Later a model of the BWR plant at Barsebäck has been implemented. In general, the degree of detail is higher at those parts of the plant that are important in accident situations.

The strategy generator is rule-based and has been written in the real-time expert system shell G2 from Genesys. It is based on the idea of safety objective trees. Following different branches of the tree, each accident case is specified in more detail.

The man-machine interface has been made in Picasso-3. It consists of a relatively small number of basic pictures and additional pop-up windows can be retrieved when needed. The same pictures that give a snapshot of the present state of the plant, are also suitable to display the future state and the past state as well.

The system has been integrated by a distributed object-oriented system called Orbix. Data is kept with the module that produces it and is not transmitted until asked for. Data is distributed, and communication is based on a client/server concept.
The simulator is running with a speed of 8 times faster than real time (HP-735). In the future the simulator will be tested in a number of scenarios, to evaluate the performance and capabilities.

The user interface can be considered as a first design, sufficient for initial testing, but it will be developed further in cooperation with end-users to fulfill their requirements.

CAMS can be used in training and during emergency exercises. The system was installed and tested in an emergency exercise in Sweden, May 1995. The purpose of the test was to evaluate how a tool like CAMS could support the analysis group during the evolution of the accident scenario.

Although the CAMS prototype has been made as a BWR version, the concept is relevant for PWRs as well and interest for development of a PWR version of CAMS has been expressed by several PWR utilities.

INSTRUMENTATION NEEDS FOR APPLICATION OF CAMS

- **Important Parameters**

Specific severe accident instrumentation, which would survive the extreme conditions in the containment, could give valuable information during accident: possible recriticality in-vessel and ex-vessel, gas concentration in the containment, pressure, temperature and water level in the containment, location of debris, temperature of the vessel lower head giving information on the risk for vessel failure, temperature of the containment basement, temperature of the core as long as it is possible before substantial core degradation.

- **Use of measured data in CAMS**

The severe accident code calculations might be correlated with the instrumentation readings, which could give a better understanding of the plant state and further progression of the accident.

Especially the containment pressurisation prediction is important regarding the timing of the possible release to the environment and the minimisation of radiation doses. Based on information of decay heat, amount of noncondensable gases in the containment and oxidation reaction heat and actual pressures and temperatures in the containment, planned CAMS calculational models will predict the containment pressurisation rate without need for more extensive severe accident modelling.

TVO already uses the ROSA system to predict the environmental effects of fission product releases to the environment. The release should be timed so that the radiation doses are as low as possible.

- **Possible Implementation of Severe Accident Codes in CAMS**

At this developmental stage the codes can give a spectrum of different possible accident progressions. With a severe accident code implemented in CAMS one would require several parallel runs during an accident with different user-input parameters to obtain that spectrum. Considering the time required for the runs and evaluation of the results, and the time available for successful accident management, it is not possible at present to base short-term accident management strategy on predictive calculations.
CONCLUSIONS

CAMS can be used to identify the initiating event of the accident to predict the accident progression until the start of core degradation with reasonably reliable results. The prediction of the core degradation and vessel failure is very uncertain at this stage which can be seen also from various analyses performed with MAAP4, MELCOR and SCDAP/RELAP5 codes. Specific plant instrumentation of severe accidents could improve the prediction. Actually, severe accident instrumentation is necessary for obtaining reliable results with CAMS. More reliable analysis for system supporting decision making could be achieved with a parallel application of severe accident codes. When the reliability of the codes has increased, application of a single code may be possible. The code results should include an uncertainty estimation, too. When the reactor pressure vessel has failed CAMS could be used to predict the pressurisation rate of the containment based on measured data of containment conditions. CAMS could also contain models to predict the environmental effects of radioactive releases, assisting in timing of the release to minimise population doses.

The accident management strategy should be based on emergency operating procedures and severe accident management guidelines. CAMS calculations made during the accident are mainly intended as support to gain more information of the possible state of the plant.

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Fig. 4. CAMS Overview Picture with status of key parameters.
Fig. 6: History and prediction of key core parameters.
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ABSTRACT

Severe accident management guidelines (SAMGs) have been developed by the three pressurized water reactor vendor owners groups; Babcock & Wilcox (B&W), ABB-Combustion Engineering (CE), and Westinghouse (W) to support the utility industry effort to upgrade their accident management capabilities. These SAMGs are considered a major step forward in providing guidance to utilities for managing severe accidents at nuclear facilities. The owners groups obviously committed major resources in the development of these SAMGs and have done so as an industry initiative.

While the SAMGS will greatly enhance accident management capabilities at the utilities, there remains unresolved issues, particularly in the area of phenomenological behaviors governing severe accidents that can introduce uncertainties into some of the decisions which must be made in responding to severe accidents. These issues are discussed so that potential side effects of implementing some of the accident management strategies can be better understood.
1.0 INTRODUCTION

The Babcock & Wilcox (B&W), ABB-Combustion Engineering (CE), and Westinghouse (W) owners groups are to be complemented for completing a major effort to develop severe accident management guidelines (SAMGs) for pressurized water reactors. These vendor owners groups have clearly expended major efforts in support of the utility industries need to upgrade their accident management capabilities. These SAMGs are considered a major improvement in providing guidance to utilities for managing severe accidents and will enhance their capabilities to respond to a wide range of severe accidents.

While the SAMGs greatly enhance accident management capabilities at the utilities, there remains unresolved issues, particularly in the areas related to phenomenological behaviors governing severe accidents. These uncertainties give rise to associated uncertainties with some of the decisions which must be made in responding to severe accidents conditions. It therefore seems prudent to discuss these issues so that the potential downsides of selected accident management strategies can be better understood. In addition, some difficulties in implementing the accident management plans may also be encountered by the utilities and these areas are also discussed.

It is hoped that the areas of uncertainty and difficulties anticipated in implementing accident management plans, enumerated in this paper, will assist the utilities in developing a better understanding of accident management. It is also very important to note that severe accidents are very low probability events. As such, in addressing the areas of uncertainty and difficulties discussed below, one should avoid expending excessive amounts of resources that would divert attention from the more probable events. The intent of accident management is to provide capabilities to deal with the severe accident consequences and in particular, the unexpected events, but not at the expense of improving our understanding and ability to deal with the more probable occurrences.
2.0 DISCUSSION

The goal of severe accident management is to enhance the capabilities of the Emergency Response Organizations (ERO) at nuclear facilities to mitigate the consequences of severe accidents, while preventing or minimizing off-site releases of radioactivity. The key objective of severe accident management is to establish core cooling and ensure that current or immediate threats to the various fission product barriers are being managed. To accomplish this, the ERO must make use of all available plant capabilities including standard and non-standard use of plant equipment and systems. The SAMGs, developed by the vendor owners groups, provide the utility guidance to achieve the above goals and objectives of severe accident management. The SAMGs would be used by the utility industry to develop plans that will enable assessment of plant damage, to plan and prioritized response actions, and to implement strategies that describe actions to be taken both inside and outside the control room. The guidance includes approaches for evaluating plant conditions and challenges to plant safety functions, operational and phenomenological conditions that may influence the decisions to implement a strategy, which will need to be evaluated in the context of the actual event, and a basis for prioritizing and selecting and appropriate strategies and approaches for evaluating the effectiveness of the selected actions.

During the assessment of plant conditions and challenges and the ultimate decision to choose a strategy to mitigate or prevent further consequences, it should be recognized that there remain some conditions which present a degree of uncertainty that could impact the effectiveness of severe accident management. The purpose of this paper is to identify these areas of uncertainty that could impact the decision making process. Areas where there may be difficulties in implementing the accident management plans are also delineated.

Areas of uncertainty that could impact the decision making process include:

1. steam generator depressurization during station blackout events
2. containment hydrogen combustion
3. very low injection rates into a highly degraded core
4. boron precipitation in the cavity
5. applicability and reliability of existing analyses of plant behavior to support the SAMGs

Areas that may cause difficulties in SAMG implementation include:

1. unavailability and reliability of instruments under severe accident environmental conditions
(2) the role of the control room during the transition from the EOPs to the SAMGs

(3) limited resources and incorporation of new information into the SAMGs

The uncertainties will be discussed first followed by the difficulties anticipated during the implementation process.

(1) Steam Generator Depressurization During a Station Blackout Event

To assess steam generator tube integrity during a core melt sequence with the reactor coolant system (RCS) at high pressure and the secondary side of the steam generators depressurized, a bounding analysis was performed for a station blackout event. The model used in this evaluation allowed the hottest temperature steam to enter the tube region from the hot legs and is contrasted with previous analyses which only simulated bulk or average conditions in the steam generators during a station blackout event. Given the uncertainty in station blackout calculations including the lack of aging effects in the failure models which will increase the tube failure potential, the Ref. 1 evaluation was performed as a bounding analysis in an attempt to show that tube failure would not occur even if the hottest steam exiting the vessel is allowed to enter the active tube region.

The results of the Ref. 1 study, however demonstrated for a station blackout event with the hottest steam allowed to penetrate the steam generator active tube region and with the steam generator secondaries depressurized, failure of the steam generator active tubes occurred prior to surge line or hot leg failure. Failure, under these assumed conditions, was not predicted in the event the secondaries remain pressurized. Fig. 1 displays the three-region steam generator nodalization used in the evaluation. Fig. 2 illustrates the tube, surge line, and hot leg temperatures for the case with the secondary system depressurized. Fig. 2 shows that creep rupture failure of the steam generator tubes can occur prior to other component failures. Also note that this secondary depressurization may be attempted to restore the secondary heat sink in the event only a non-standard feedwater pump is available which can only operate at very low back pressures. This desire to depressurize the RCS using the steam generators may be motivated by the need to reduce RCS pressure since only low pressure emergency core cooling (ECC) injection may be available for core cooling. Activating low pressure ECC injection would be needed to attempt to maintain the core within the vessel.

The importance of these results suggest that secondary steam generator depressurization should not be attempted following a station blackout event when the loops contain highly superheated steam since the actions could produce multiple failures of the active tubes. In lieu of the potential for tube failures, the additional uncertainties associated with condensation of primary steam, in the presence of large quantities of noncondensable gasses in the RCS, may not facilitate the desired RCS depressurization. As such, the actions to depressurize the steam generators may produce failed tubes without
achieving the eventual addition of low pressure ECC to cool a relocated core and prevent vessel failure, which is the desired result.

It should also be noted that a review of other operator actions is currently in progress under a follow-up program to assure that other strategies would not change the conclusions of this nor previous studies investigating steam generator tube failure potential during severe accidents. These additional analyses are currently being conducted to assess the effects of nodalization, other operator actions, and aging on the potential for creep rupture failure following the station blackout event.

Also note that these analyses are consistent with the cautions in the Westinghouse SAMGs regarding secondary depressurization.

(2) Containment Hydrogen Combustion

Hydrogen combustion can pose challenges to the containment through static (deflagration and diffusion flame) or dynamic (detonation) overpressurization, missile generation, and equipment failure resulting from thermal or pressure effects. Diffusion flames and slow deflagrations are not expected to represent a serious threat for most containments. Therefore, the major concern is directed toward accelerated flames and the transition to detonations. As discussed in Ref. 3, the key parameters that affect the likelihood of a detonation are the geometry and thermodynamic state of the mixture. Many containments are complex and the thermodynamic state is highly accident dependent (stratified conditions versus partially mixed). As such, the distinction of the boundaries between the flammability and detonation limits can have very dramatic variations. The assessment of the likelihood of a burn or detonation requires knowledge of the hydrogen concentration in the containment and the flammability limits for a steam-air-hydrogen mixture. The SAMG models used to develop the flammability and detonation limit curves are derived based on simple models and idealized conditions where uniform mixing is assumed. While simple models can be used to derive these limits in a well mixed atmosphere, uniform mixture conditions cannot be assured, leading to much uncertainty in the calculated limit boundaries. As a consequence, use of the flammability limit curves in accident management could give rise to unexpected deflagrations and/or detonations. Since the distribution of hydrogen in the containment remains unknown and given the uncertainty in the limit curves derivation, management of containment hydrogen combustion is expected to be uncertain. This uncertainty in being able to manage the potential for containment hydrogen combustion should be emphasized since the use of the SAMG flammability/detonation limit curves could present the accident management team with a "false sense of security" regarding when burns or detonations would really be expected. To account for these uncertainties, the licensees should modify the flammability and detonation limit curves to include error bars to indicate the plant specific variation in hydrogen concentration that could occur as a result of localized discharge of steam hydrogen mixtures into the containment, the accumulation of hydrogen in certain compartments or operator actions to spray the containment in an attempt to maintain inerted containment conditions.
(3) Injection During Highly Degraded Core Conditions

The SAMGs provide analysis aids to determine, for example, the injection rates to cool highly degraded cores. However, there is little guidance as to what the course of action should be, if only small rates of injection are available. This is a plant specific issue since the flow rates from various pumped systems and the injection locations differ significantly among plants. The concern is that while a much lower rate of injection will not quench the core, this lesser flow will simply produce hydrogen and not contribute to cooling. The TMI-2 accident certainly suggests that there are benefits of having water in the vessel to cool a core that relocates. As such, a very low injection rate into the cold side of the system would fill the lower plenum and possibly preclude lower head failure following a relocation. On the other hand, if only injection into the hot side of the system is available (i.e., remember plants have hot side injection capability and pressurizer spray presents a path for hot side water addition), the low injection rate into the core from the hot legs would simply exacerbate the accident which could cause the accident management team to question the actions.

As part of developing the plant specific SAMGs, licensees should structure their guidance to deal with the situation described above and provide appropriate cautions.

(4) Boron Precipitation in the Cavity

This issue will arise when the core has left the vessel and occupies the cavity. For those plants that can flood the cavity, boiling of the relocated core could persist for many hours. Under these conditions, the boron acid could reach solubility limits and precipitate. The loss of heat removal capability from the relocated core materials as a result of the crystalline precipitate may re-initiate melting and the further degradation of the cavity concrete and basemat materials. This issue is raised because there is no information in the literature treating the consequences of precipitation in the cavity that could occur during the long term when the core has exited the vessel. It is expected that the precipitate would eventually melt but no evaluations of the long term coolability have been documented. Perhaps some future studies could be performed to show that boron precipitation does not pose a long term cooling concern in the event sufficient water supplies become available in the later stages of an event to cool a core that has relocated into the cavity. The potential uncertainties could cause problems during plan implementation or again pose many concerns to an accident management team during an accident.

(5) Applicability and Reliability of Existing Analyses of Plant Behavior to Support the SAMGs

A multitude of existing analyses can be utilized to support the development of the SAMGs. Individual Plant Evaluations (IPEs), for example, constitute a major portion of the analyses performed to identify plant vulnerabilities and capabilities under severe accident conditions. Some IPEs, however, appear to have utilized extremely simplified thermal-hydraulic codes to identify the
minimum set of equipments for successful core cooling and have not been verified against the results of the more detailed thermal-hydraulic calculations. In the event accident management strategies are derived from these results, the strategies may not achieve the desire plant response. It is therefore recommended that, in the event strategies are devised based on IPEs utilizing simplified methodologies, the IPE thermal hydraulic analyses be carefully reviewed to assure the code has been properly benchmarked and correctly exercised so as to produce a technically sound and consistent result. Hand calculations can easily be performed to check some of the code results such as checking that primary and/or secondary relief valves are relieving coolant at the correct flow rates and that the level swell (and void distribution) in the core is consistent with the core decay heat steaming rate, power distribution, and ECC injection flow. Other plant calculations could also be surveyed to see if accident simulations exist that could be used to judge the IPE simplified model results. These many checks should be performed to assure that the margins of safety remain at their maximum and the IPE plant vulnerabilities and capabilities assessment, if incorrect due to calculational errors, does not also infect the severe accident management plans.

Some of the difficulties that may be encountered during implementation of the SAMGs include:

1. Unavailability and Reliability of Instruments Under Severe Accident Conditions

In order for accident management to be effective, the emergency response organizations will need to correctly interpret the many instrument responses. Many decisions regarding strategies affecting in-vessel and containment performance will rely on the ability of a select group of instruments to provide useful information. When conditions begin to exceed the qualification of the instruments, instrument error increases until ultimately the instrument may fail. Moreover, prior to failure, the severe accident environment may cause instrument error to become very high. While the magnitude of a measured parameter may be so high that it may no longer be reliable, its trend may still be valid. It would therefore be useful to identify instrument behavior, for those select group of vital instruments identified as critical to effective accident management. The range of environmental conditions such as pressure, temperature, and radiation should be identified for the case when the magnitude of the measured parameter becomes severely impaired but the trend is correct, and for the case when the trending ability of the instrument becomes invalid. Thus, instruments with unreliable parameter magnitude indications may still give important information because it can still properly trend the performance of the measured parameter. For example, although an instrument parameter magnitude is erroneous, its trend for decreasing pressure or temperature could assure the response team that a given strategy is having the desired beneficial effect. Identification of the erroneous magnitude and reliable trend boundaries for instruments during severe accident conditions would provide additional information to the response teams. This information is considered very important because the success of accident management depends, for the most part, on the ability of the response team to gather the
correct information and make decisions based on the key instrument responses. This critical information would probably come from a small select group of instruments at the facility, so the effort to collect this information is not expected to be an overwhelming exercise.

(2) Role of the Control Room During the Transition From the EOPs to the SAMGs

The SAMGs provide guidance for the control room on their role and actions when severe accident conditions develop prior to staffing of the Technical Support Center (TSC). Moreover, the extent of this guidance varies among vendors. Although specific severe accident guidelines may not be provided to the control room operators, they are expected to be familiar with the TSC guidance. The familiarity with the role of the TCS can be gained through operator training and participation in severe accident management drills. These drills should be designed to assure that the control room staff will take appropriate actions up to the time at which the decision making responsibility is transferred to the TCS. This suggestion is made in light of the fact that many utility TSCs may not be fully staffed and operational for as long as two hours following an activation. This focus on the transition from the EOPs to the SAMGs during drill exercises could ensure that the control room personnel know their roles and responsibilities should a need to transition out of the EOPs be required when there is no support from the TSC.

In addressing this issue, it must not be forgotten that the operators cannot be overburdened with an excess of responsibilities. Their roles must be kept in perspective, as an excessive training program could lead to a dilution of the operators’ primary role in dealing with anticipated or expected minor events. Use of drill exercises that emphasize the operators’ role during the transition period should reduce the need for extensive training programs in severe accident behavior for plant operators.

(3) Limited Resources and Need for Incorporation of New Information

Some utilities may have limited resources and capabilities regarding the understanding of severe accidents and its many attendant issues. This is not a condemnation of the smaller utilities, rather it is recognized that utilities with a single nuclear facility can not justify the staffing levels and accident analysis capabilities of those with several units. As a consequence, some utilities may encounter difficulties in developing plant specific accident management guidelines from the generic SAMGs and implementing these guidelines at their facilities. Accordingly, it is anticipated that the owners groups and consultants will remain active in assisting those utilities in implementation phase to assure all utility members have acceptable severe accident management plans in place. In this same light, it is further suggested that the owners groups provide periodic input to the utilities to maintain the most up-to-date plans. That is, as new information becomes available, the vendor owners groups could provide a consistent set of recommendations in regards to translating the latest research information into the utility severe accident management plans.
Lastly, it may be instructive to mention that the SAMGs are based on accidents from full power conditions. Accidents from low power or shutdown have not been addressed in the SAMGs. Moreover, it is felt that the knowledge and capabilities in dealing with accidents from full power should allow one to deal equally with accidents from shutdown. The overall objectives of adding water and cooling the core obviously remain the same, however equipment availability may differ and thereby preclude the use of some strategies that were designed for accidents from full power operation. During shutdown, injection systems in particular, are removed from service and pose special considerations that may not have been allowed for in the strategies devised to cope with accidents from full power. Again, the intent here is not to suggest a major expenditure of resources and effort to cover low power or shutdown conditions. Rather, with the recent completion of an abundance of plant specific information regarding accidents from shutdown, use of this existing information and knowledge, where feasible, is recommended for inclusion in the SAMGs. In the event it is not feasible to include shutdown accidents at this time, then it is suggested that this information be considered for future upgrading of the plant specific SAMGs. Guidance for accident from shutdown is considered important since during the low power and shutdown modes of operation, systems are removed from service limiting the use and timeliness of many strategies that would otherwise be available to the accident management teams following accidents from full power operation.
3.0 CONCLUSIONS

The vendor owners groups have committed major resources to creating SAMGs for use by the utility industry in improving their accident management capabilities and are commended for their efforts. The SAMGs represent a major step forward in expanding the ability of the utilities to cope with severe accidents at their facilities.

In the management of severe accidents, there remains, however, uncertainties and unanswered questions regarding some aspects of plant damage state diagnoses and the decisions to implement strategies to mitigate or prevent further accident consequences. These areas where uncertainty that could impact the decision making process include:

(1) Steam generator depressurization during station blackout events could lead to steam generator tube ruptures which suggests that secondary depressurization should not be attempted in the event highly superheated steam is entering the external loops.

(2) Containment hydrogen combustion may be difficult to effectively control due to the simplistic models used to define flammability and detonation limits. Since the hydrogen distribution is also not known in the containment, the ability to manage the containment hydrogen combustibility will be plant and sequence specific and will include the need to deal with the attendant uncertainties.

(3) Injecting ECC water at low rates into the hot side of the RCS or directly into a highly damaged core could produce hydrogen and exacerbate the accident consequences. The negative aspects of low injection rates could pose concerns for the accident management teams and should be considered in the accident management planning stage to avoid complications during the actual event.

(4) The injection of borated water into the cavity when the core has exited the vessel could result in boron precipitation that could challenge debris coolability. Additional studies should be performed to address the consequences of boron precipitation during the long term cooling of the core debris in the cavity.

(5) If simplified IPE thermal hydraulic analysis results are used to directly support development of accident management plans, the IPE analyses should be carefully reviewed to assure the IPE results are not translated into incorrect accident management strategies.

Some of the areas where some difficulties in implementation of the SAMGs into severe accident management plans may be encountered and include:

(1) Instrument error under severe accident conditions has not been identified in the SAMGs. Since the ability of the accident management team to effectively manage accidents is dependent upon
the instrument response, it is recommended that the behavior of the key instruments used in the management of accident be evaluated. This should include the environmental conditions where the instrument measured parameter is no longer valid but the trend remains correct, and the conditions where the trend is no longer correct due to harsh environment. Thus, although the instrument cannot measure the magnitude of the sensed parameter accurately, the correct trend can provide useful information to the accident management team.

(2) Drill exercises should be used as a vehicle to train the operator control room staff in their expected role in the event a transition from the EOPs to the SAMGs is required prior to staffing of the TCS.

(3) A lack of a comprehensive knowledge of severe accident phenomenological behaviors could make implementation of the plant specific SAMGs at some utilities difficult. The vendors and other contractors are expected to provide assistance to those utilities which cannot support large technical staffs. The vendor owners groups can also provide evaluations of new research information as it becomes available to assure the severe accident management plans maintain the most up-to-date information.

The SAMGs are formulated based on accidents from full power. Accidents from shutdown can pose additional considerations during the management of accidents. As such, guidance for accidents from shutdown is also considered important since during the low power and shutdown modes of operation, systems are removed from service. Removal of many of these systems from service can limit the use and timeliness of many strategies that would otherwise be available for use by the accident management teams in the event of an accident from full power operation. Since there has been an abundance of information published regarding accidents from shutdown, it is recommended that this information be considered for inclusion into the plant specific SAMGs.

It is hoped that the areas of uncertainty and difficulties anticipated in implementing the accident management plans that are discussed in this paper will assist the utilities in the developing effective accident management plans at their facilities. In some cases, where there may be no clear answers, however, the awareness of such uncertainties and the consideration of the negative aspects of all of the strategies may at least minimize errors or preclude unnecessary decisional delays during the management of severe accidents at nuclear facilities.

It is also very important to note that severe accidents are very low probability events. We must, therefore, not also forget that the intent of accident management is to provide capabilities to deal with the severe accident consequences and in particular, the unexpected events, but not at the expense of improving our understanding and ability to deal with the more probable occurrences.
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Fig. 1. Three Region Steam Generator Nodalization.
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IN PURSUIT OF
CONSISTENCY AND COMPLETENESS IN THE
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ABSTRACT

An approach has been developed to complete defense-in-depth strategy with the severe accident assessments and management in order to meet the deterministic Finnish criteria concerning severe accidents. The approach includes a number of hardware changes and procedures. The status of application to the Loviisa VVER-440 plant is presented. Consistency and completeness of the approach is discussed in order to demonstrate that a closure of defense-in-depth principle may be reached. This requires a probabilistic treatment to screen out less likely sequences, quantification of severe accident phenomena and treatment of uncertainties at all levels. Application of the gained experience to other concepts is highlighted.

1.0 INTRODUCTION

The approach adopted to severe accident management (SAM) at the Loviisa nuclear power plant in Finland was presented and discussed in the previous OECD/CSNI Specialist Meeting on SAM Programme Development organized by SESAM Group in 1991 [1]. The approach was called consistent, since it is mechanistically based, and the overall aims and methodology are risk-oriented. Thus it combines the deterministic and probabilistic aspects, having a particular emphasis on phenomenological uncertainties. The plant-specific SAM approach to the Loviisa allows us to reduce, and in many cases even to eliminate these uncertainties.

The focus of the Finnish requirements, according to Regulatory Guide YVL 1.0 (1982) and YVL 2.2 (1987), is to maintain the containment integrity during severe accidents, having only a remote likelihood of the failure. Additionally, no acute radiation damage is allowed to nearby inhabitants, and long-term larger land and sea contamination is to be prevented. The required measure for preventing contamination is that maximum release of cesium-137 is specified to be lower than 100 TBq [2]. When applying these requirements to the plant, IVO translated them to mean that the containment failure has to be a very unlikely event. The conceptual scale to quantify the likelihood was already discussed in detail [1]. Accordingly, the severe accident management goal is that the conditional containment failure probability of less than $10^{-2}$ should be demonstrated for each major class of severe accident sequences. For
the quantification of each physical failure mechanism, the Risk-Oriented Accident Analysis Methodology (ROAAM) [3] is applied.

During the years 1990-1993 extensive feasibility studies were conducted by Finnish nuclear power utilities to study the feasibility of the new nuclear units offered to Finland by several vendors. All the offered plants represented an evolutionary version of existing PWR's and BWR's. The vendors were required to demonstrate that the Finnish design requirements can be fulfilled by applying basically the similar approach that will be discussed in Section 2. Application to Russian NNP-91 concept of VVER-1000 has been discussed by Antropov in this Specialist Meeting [4].

Section 3 presents the SAM application and implementation status at Loviisa. Successful severe accident prevention is demonstrated with the level 1 PSA studies. Low fraction of sequence classes leading directly to large releases is to be shown with separate studies. The treatment of severe accident scenarios is based on the top level critical SAM safety functions. Part of the required measures to satisfy the SAM safety functions has been implemented and the rest are still in the investigation and planning phase.

Consistency and completeness of the approach will be further discussed in Section 4. The success criteria are defined for the different levels of treatment. The role and treatment of uncertainties during quantifications will be discussed briefly.

2.0 STRUCTURE OF THE APPROACH

The structured approach is based on the successive levels of assessment and management, as summarized in Table I. The prerequisite for the severe accident assessment and management (SAAM) work is that prevention of the severe accidents has been attended to the degree practically achievable. The primary purpose of the SAAM is to achieve an effective mitigation of large releases. The only way to do this is by ensuring the containment integrity throughout a severe accident. Thus, the focus is in the containment integrity. The containment isolation has to be successful, the initiator should not lead to bypassing the containment, the energetic phenomena should not cause a catastrophic failure of the containment, the temperature and pressure buildups should not make the containment leak.

Accordingly, the first level is to show that all practical measures have been taken to prevent the accidents. The measure is that the full-scope PSA level 1 shows a very low frequency of core damage sequences (CDF). For the existing plants we apply the target value of $10^{-4}$/r-yr, while for the new plants the CDF target value is set to $10^{-5}$/r-yr.

The second level is to show the very low fraction of those core damage frequencies that can be assumed to lead directly to large releases. Such core damage sequence classes are the sequences with impaired containment system function, high-pressure sequences and reactivity initiated core damage sequences. We set a target value for the fraction of each of these sequence classes to be lower than $10^{-2}$. Certain flexibility has to be accepted for the fraction goals in case that the CDF is shown to be considerably lower than its target value.
**TABLE I** The successive levels of the SAAM

<table>
<thead>
<tr>
<th>Level</th>
<th>Description</th>
</tr>
</thead>
<tbody>
<tr>
<td>I</td>
<td>Level 1 PSA</td>
</tr>
<tr>
<td>II</td>
<td>System reliability: low fraction of</td>
</tr>
<tr>
<td></td>
<td>- sequences with impaired containment system function</td>
</tr>
<tr>
<td></td>
<td>- high-pressure sequences</td>
</tr>
<tr>
<td></td>
<td>- reactivity initiated core damage sequences</td>
</tr>
<tr>
<td>III</td>
<td>Absence of containment failure due to physical phenomena</td>
</tr>
<tr>
<td></td>
<td>- energetic events: in-vessel and ex-vessel steam explosions, hydrogen burns, DCH, missiles</td>
</tr>
<tr>
<td></td>
<td>- slow overpressurization: steaming and generation of noncondensable gases</td>
</tr>
<tr>
<td></td>
<td>- core-concrete attack</td>
</tr>
<tr>
<td></td>
<td>- induced steam generator tube failures</td>
</tr>
<tr>
<td></td>
<td>- recriticality of the degraded core and core debris</td>
</tr>
<tr>
<td></td>
<td>- temperature loadings of the containment</td>
</tr>
<tr>
<td>IV</td>
<td>Radioactive releases through containment leakages</td>
</tr>
</tbody>
</table>

**IMPLEMENTATION LEVEL**

Severe Accident Management Concept including
- hardware, instrumentation and information needs
- procedures
- organisational implementation

On the second level of system reliability, the wide class which we have named the **sequences with impaired containment system function**, consists of

- by-pass sequences, where the initiating event is a leak outside the containment through an interfacing system (the most important being primary-to-secondary side leakages);
- sequences with pre-existing openings;
- containment isolation failures;
- sequences by-passing the containment pressure suppression system and thus leading to the containment failure before the core melt starts, and
- sequences with an induced leakage outside the containment (e.g. ECCS suction line failure, recirculation failure to pump the coolant from sump to the ECCS tank).
A large variety and versatility of these sequences causes difficulties in the definition and quantification. Particularly these sequences tend to be extremely plant-specific.

The reason to require a low fraction of high pressure sequences relates to the severity of the high pressure melt ejection for the containment, and to the possibility of the induced steam generator tube failures. On the other hand, the depressurization is also crucial from the prevention viewpoint, since primary side bleed&feed would be efficient to stop the accident progression. Thus depressurization makes a natural interface between preventive and mitigative phase of SAM. If primary feed is successful, the accident progression will be stopped. If primary feed is not successful, depressurization is to be understood as the first mitigative action, and operators would turn from the core cooling oriented actions to mitigate the consequences.

The reactivity initiated core damage sequences, where the excessive core neutron power causes the core damage, should be omitted. The consequences of such sequences might be very serious or they might be subject to large uncertainties. The recent experiences show that nonhomogeneous boron dilution accidents could be such sequences in the PWR’s. Some anticipated transient sequences without scram could pose also reactivity problems in the BWR’s.

After demonstrating the containment availability from the system related viewpoint, the efforts should be directed to show absence of containment failure due to the physical phenomena. It should be shown that any physical phenomena do not challenge the containment integrity during the accident. The phenomena to be studied include in-vessel and ex-vessel steam explosions, hydrogen burns, DCH, missiles, slow overpressurization due to steaming and generation of noncondensable gases, core-concrete attack, recriticality of the degraded core and core debris, and temperature loadings of the containment. Again here, it should be remembered that plant-specific studies are necessary for a proper treatment of phenomena and associated uncertainties.

The difficulty of risk-oriented treatment of physical phenomena are strived to be overcome with applying the ROAAM approach in cases where the containment failure cannot be directly excluded.

After successful exclusion of the containment system and structural failures, the final level is to define the radioactive releases through containment leakages. The releases during the managed accident sequences should stay below the acceptance criteria concerning acute health effects and land contamination. E.g. when considering the Finnish acceptance criterion 100 TBq of Cs-137 release, the corresponding acceptable leak size for the PWR’s tends to be one order of magnitude higher than the design leakage of 0.1% a day.

Depending on the individual plant or plant concept, ensuring most of the above levels require certain implementation measures to be taken. We call all these measures as a SAM concept. This concept includes all the hardware and instrumentation measures, what are needed to achieve the set targets in levels I through IV. The operator information needs have to be defined, and they are to be implemented in line with development of the SAM guidance and
procedures. The organizational structure and responsibilities during the accidents have to be defined.

3.0 APPLICATION AND IMPLEMENTATION STATUS AT LOVIISA

A number of specific features characterizes the Loviisa plant. Two units of Soviet type VVER-440 reactors have been equipped with ice condenser containments. The ice condensers were constructed under a Westinghouse license, but they have some crucial differences in comparison to the other 12 ice condenser containments in the world.

The overall severe accident management approach was structured around the idea of demonstrating in-vessel melt retention, hydrogen management and reliable long-term containment cooling. The in-vessel retention concept was first introduced to Loviisa assessments by Theofanous [5]. The effective management of severe accidents requires also depressurization of the primary circuit to ensure low pressure during the long-term contact of the molten corium pool with the reactor vessel.

Accordingly, the top level critical functions of the Loviisa SAM scheme are defined as

- depressurization of the primary circuit;
- absence of energetic events, i.e. hydrogen burns;
- coolability and retention of molten core on the lower head of the reactor vessel, and
- long-term containment cooling.

Additional efforts are needed to ensure the containment integrity against impaired system functions.

Ensuring of these safety functions is the main task of SAAM being performed for the Loviisa plant. The SAAM tasks have involved phenomenological studies, development of the SAM strategy, necessary backfitting measures (including hardware and instrumentation) and writing the SAM guidance for the personnel.

3.1 PSA STUDIES AND CONTAINMENT SYSTEMS' RELIABILITY

The full-scope level 1 PSA for Loviisa Unit 1 has been carried out, and currently "Living-PSA" is routinely used for various purposes. The initial results were obtained for internal events in 1989, and since then several updates have been made after a number of hardware and procedure modifications, as discussed by Mohsen and Vaario [6]. The estimated core damage frequency will be $4 \cdot 10^{-5} /r$-yr (after implementation of the decided modifications) [7]. Currently, external events have been added and shutdown conditions are being worked out. Overview of plant damage states and grouping in the accident management perspective was discussed before in Ref. 1.

The work has been initiated for defining the core damage sequences with impaired
containment system function. At the first glance, the total fraction of by-pass sequences (particularly large primary-to-secondary side leakage accidents) and sequences with induced leakage outside the containment seems to be unacceptably high. Thus a thorough and truly plant-specific study was deemed necessary for defining acceptance criteria, grouping the sequences correspondingly and quantifying their importance. Previously decided plant modifications to allow better management of the primary-to-secondary leakage accident are under construction [8]. Elimination of significant core damage sequences leads to lower CDF values, and it is difficult to fulfil a criterion that the fraction should be less than $10^2$. Higher fractions can be considered acceptable, if the management frequency window $10^{-6} - 10^{-4}$ per reactor year is not seriously violated.

A low fraction of high pressure sequences will be ensured with intentional primary circuit depressurization as discussed in next section.

Main contribution of reactivity initiated core damage sequences comes from boron dilution events. Dilution sequences from external sources have been incorporated into PSA level 1 sequences. Inherent boron dilution mechanisms during accident sequences (boiling-condensing mode during LOCAs, ATWS, primary-to-secondary leakages etc) are currently being investigated in detail.

3.2 ENSURING TOP LEVEL CRITICAL SAFETY FUNCTIONS

Primary circuit depressurization

The primary depressurization is an interface action between the preventive and mitigative parts of SAM. If the primary feed function is operable, the depressurization will prevent the core melt. If not, it sets in motion the mitigative actions and measures to protect the containment integrity and mitigate large releases.

The decision has been made to install manual depressurization capability by motor-operated relief valves. Depressurization capacity will be sufficient for bleed&feed operation with high-pressure pumps, and for reducing the primary pressure before the molten corium degrades the reactor vessel strength. Depressurization is to be initiated from the first indications of superheated temperatures at core exit thermocouples. The installation of the depressurization valves is planned at the same time as replacing the existing pressurizer safety valves in 1996. The initial idea of implementing depressurization by manual operation of the safety valves was abandoned mainly because of difficulties in the valve qualification.

Absence of energetic events

It is possible to demonstrate absence of energetic reactor vessel failure and absence of vessel melt-through. Thus, the only real energetic concern is due to hydrogen combustion events. Because of the relatively low design pressure, this concern involves all large scale combustion events that are rapid enough to yield an essentially adiabatic behavior. Glow-plug igniters
were installed in the Loviisa containments in 1982. An extensive research program has been conducted at IVO since 1988 to study the reliability and adequacy of the existing igniter system. The work includes the experimental program with the VICTORIA facility and associated development of the calculational models. These studies are now near to the completion [9]. The new hydrogen management scheme will concentrate on two functions: ensuring air recirculation flow paths to establish well-mixed containment atmosphere (opening of ice condenser lower doors) and effective recombination and ignition.

**Lower head coolability and melt retention**

In-vessel retention of molten core on the lower head of the reactor vessel constitutes the cornerstone of the Loviisa accident management approach. Since the ice condensers melt in most accident scenarios, the reactor cavity is filled with water and the reactor vessel is submerged. A typical decay heat power level is low (~9 MW). Thus local heat fluxes and vapour velocities around the vessel are rather moderate. As an accident management measure, however, the lower head insulation and neutron shield blocks should be lowered during the accident. The results of the study (see Ref. 10 for details) have been submitted to Finnish regulatory authority STUK for the scrutiny and approval. The implementation of the necessary measures at the plant is foreseen in years 1997 - 1998.

**Long-term containment cooling**

In the absence of corium-concrete interactions, there is no production of non-condensible gases, except hydrogen which is to burn. Stabilization of containment pressurization can be achieved by steam condensation on the containment walls. Thus, an external spray system of the inner steel containment can replace a filtered venting system that was originally planned for Loviisa after the Chernobyl accident.

External spray systems were constructed on the Loviisa containments in the years 1990 -1991 with the design objective to prevent slow overpressurization due to continuous steaming during severe accidents [11]. The associated long time delays allow utilization of an active system. An extra benefit is that deliberate release of noble gases can be avoided. The design calculations were verified with almost full-scale and real condition experiments in the German HDR containment.

**3.3 INSTRUMENTATION, OPERATOR INFORMATION NEEDS AND PROCEDURES**

The safety parameter display system has been installed to the Loviisa, and it is supporting together with the general symptom-oriented emergency procedure the operator in his actions during the accidents. These aids are particularly helpful for the preventive actions.

For the mitigative SAM measures, new instrumentation is necessary. The planned SAM
instrumentation includes only those which are of crucial importance for performing a correct SAM measure and for monitoring the success of the measure. The new instrumentation is rather strictly limited to those helping to ensure the critical SAM safety functions, as discussed in Ref. 12. The design objective is to keep it independent of the DBA instrumentation. All SAM instrumentation have at least two trains. Also in some cases, functional and/or component diversity can be employed. The aim is to have clear technical solutions employing reliable and passive (to the possible extent) components. Special attention has to be paid on equipment located in the rooms where environmental conditions of a severe accident can appear.

The operator procedures are developed concurrently with the implementation of the hardware changes. The procedures are planned on two levels. A Severe Accident Handbook will be written for the operators and the technical support personnel, describing the expected accident progression and the management actions available. Based on this information the personnel will be able to take decisions of the needed actions, which then will be described by the operating instructions of the particular SAM equipment.

4.0 COMPLETENESS AND CLOSURE OF SAM

Different countries apply different approaches to manage the beyond DBA and severe accident situations. The Finnish Regulatory Authority has launched quite deterministic SAAM requirements [2]. When responding to these requirements, the probabilistic considerations have played a key role. The real benefit for the nuclear safety can be expected only, when treatment is consistent and comprehensive. In this section an attempt is made to show that it is a feasible objective to strive for the closure of the defense-in-depth principle by an effective SAM approach. As an outcome, it can be demonstrated that addition of severe accidents to the considered scope of accidents can complement the original DBA approach, and that a well-defined and satisfactory safety goal can be defined.

Another approach has been taken in some other countries. E.g. the USNRC requires that the licensee develops an accident management and severe accident mitigation program to further reduce large releases by applying existing equipment. The main reason, why such an approach was not taken by the Finnish organisations is that purely probabilistic approaches are not deemed sufficient. Particular goal is to ensure the safety (i.e. ensuring low consequences) to practically achievable degree. This in turn would promote the public acceptance of nuclear power in the country.

4.1 CONSISTENT AND COMPREHENSIVE APPROACH

The original defense-in-depth concept included the safety levels up to the design basis accidents. For these accidents cutoff was made by applying deterministic requirements. The role of probabilistic safety goals to be met with PSA level 1 studies is to complement the deterministic approach by demonstrating the adequacy and applicability of the DBA approach
to prevent efficiently severe accidents.

After Three Mile Island and Chernobyl accidents, it became evident that a new level should be added to prevent beyond design basis accidents (BDBA) from leading to severe accidents and to mitigate their consequences. Addition of the new level is interpreted to be fulfilled with SAM.

Different SAM approaches are taken in different countries. When we developed our approach, the starting point was to meet the deterministic Finnish criteria. Thus the requirements concerning SAM are to prevent core damage with all practical means and to mitigate the consequences by

- maintaining the containment integrity,
- preventing acute health effects, and
- preventing land and sea contamination.

The implementation principles were then defined in order to meet the requirements. Consequently, the preventive and mitigative measures include

- existing means,
- backfitting the plant when deemed necessary, and
- developing severe accident management guidelines and appropriate procedures.

When these principles were turned to practical measures, the structure presented above in Section 2 in Table I was obtained. In the practical application it is of crucial importance to define the interface between the preventive and mitigative part. In the Loviisa case, it is quite obvious that the interface is defined by the success or unsuccess of the primary system depressurization to allow water injection into the reactor core.

In order to show effectiveness of the chosen approach, the success criteria have to be defined for the different levels. In the following the applied criteria are discussed briefly.

**Prevent core damage with all practical measures.** The full-scope PSA level 1 shall show a very low frequency of core damage sequences (CDF). The full-scope PSA should include normal operation and shutdown conditions, internal and external events, and certain "spin-off" situations such as inherent boron dilution during accident conditions, boron crystallization in the core and pressurized thermal shock of the reactor pressure vessel. For the existing plants we apply the target value of $10^{-4}$/r-yr, while for the new plants the goal is set to $10^{-5}$/r-yr.

**Prevent sequences with a direct challenge to containment integrity.** The criterion is to show a very low fraction of those core damage frequencies that can be assumed to lead directly to large releases (sequences with impaired containment system function, high-pressure core melt sequences and reactivity initiated core damage sequences). A target value for the fraction of each of these sequence classes to be lower than $10^{-2}$ has been applied for the new plant designs. In case of the existing plant the studies are under way, and the practical safety goal may need to be adapted to the accident management frequency window. Also the new
plant concepts may be able demonstrate extremely low CDF values. In such cases the fraction $10^{-2}$ would lead to practically nonfeasible requirements, and adaptation will be needed.

**Prevent containment failure by physical phenomena in significant severe core damage scenarios.** The conditional likelihood of the containment failure is defined to be less than $10^{-2}$. This figure expresses that it is physically outside the spectrum of reason to assume that the given phenomena would fail the containment. Since the traditional PSA level 2 has turned out to be not sufficiently traceable, particularly not for intangible physical parameters, we have selected for quantification purposes the ROAAM method. This method produces directly quantification for the considered phenomena in a structured and traceable way. Furthermore, it allows to bring new information, when available, to quantification.

**Restrict the radioactive releases by effective SAM.** For the effectiveness of SAM measures, the acceptance criteria come from the requirement of no acute health effects and of meeting the 100 TBq release limit of Cs-137.

After successful fulfilment of these quantified criteria, the original safety goal of CDF being less than $10^{-3}$ — $10^{-6}$ /r-yr can be complemented with the statement that likelihood of large releases is two orders of magnitudes lower for each significant class of sequences. In other terms, it has been shown that assumption of large release given a severe accident is not physically reasonable. Thus large releases can be expected only in residual sequences with frequencies less than $10^{-5}$ — $10^{-6}$ /r-yr.

### 4.2 UNCERTAINTIES

The basic idea of our approach is to define deterministic SAM concept, where probabilities are used for screening and grouping the sequence classes. In principle, it is quite evident that the approach can be created in a consistent manner. However, if the uncertainties are too large the consistency of practical application can be lost.

When dealing with very low probability figures, another big question is, whether the treatment is complete, i.e. anything significant has not been forgotten.

The first source of uncertainties is in the probabilistic treatment. The level 1 PSA attempts to deal with statistically based quantifications. However, already the baseline study is bound with large uncertainties concerning initiator frequencies of rare events, and treatment of human errors and common cause failures. The associated error sources are quite different. Consequently, both absolute and relative frequency figures have large uncertainty bands, and all interpretations should be made bearing that in mind. Another difficulty might arise from rather different statistical base for internal and external events. A successful application of the PSA level 1 results requires that the analyst knows the sources of uncertainties and is able to avoid wrong decisions. One way to obtain understanding of significant contributors is to treat accident sequences separately from initiators and leaving human factors unattended. The applied event tree methodology allows a structured way to evaluate accident progression. This helps in identifying real deficiencies.
In addition to the baseline study, there are several spin-off conditions (often forgotten) which should be treated separately. Pressurized thermal shock to the reactor pressure vessel, inherent boron dilution during accidents, boron crystallization in the core, sump clogging or fuel assembly clogging with debris are examples of such spin-off conditions, which may turn a successfully managed accident sequence into a core damage sequence. For Loviisa, we have performed or are performing separate studies to quantify importance of such situations and bring them to the PSA context; see e.g. [13], [14].

On the level 2 PSA the probabilities are required for complex physical phenomena, which are not compatible with statistical quantifications. The strength of the ROAAM method in comparison with more traditional PSA level 2 methods is in treating the phenomenological uncertainties per se. The treatment is decomposed to the level where quantifications can be made based on the existing physical knowledge. Whenever expert judgements are needed, decomposition allows a focussed approach. What is more important, all new information can be easily added to the treatment when available.

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Severe Accident Management at Ringhals PWR - Present Status and Future Work

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Abstract: In 1988 measures to mitigate the consequences of severe accidents were implemented at all nuclear power reactors in Sweden. They consisted of a filtered vent containment system and back-up sources to the containment spray system. The requirement of these measures was to reduce the releases to less than 0.1% of the core inventory of fission products (for a 1800 MWt reactor) leading to limited land contamination if a severe accident would occur.

As part of the mitigating programme, new Emergency Operating Procedures (EOP) were developed. For the PWR's the Emergency Response Guidelines (ERG) were complemented with BERG (Beyond ERG), in which the main emphasis for the operator was to protect the containment.

In separate projects, a knowledge based handbook for severe accident management has been developed, and long-term effects of a severe accident have been investigated.

In this paper the present status of the severe accident management for Ringhals PWR will be presented. Furthermore, and more important, it is discussed how it may be improved. In the paper the importance to continue the work with severe accident management issues is stressed. Such work gives the mental preparedness that an accident, although unlikely, may occur. This is a safety enhancement in itself.

1. BACKGROUND

In 1986 the Swedish government issued a requirement to the Swedish utilities to implement a program of severe accident mitigation measures before the end of 1988. The basic guidelines and criteria, which by the government decision shall apply to the severe accident mitigation measures of Swedish nuclear plants are:

* The same basic requirement regarding the maximal quantity of released radioactive substance shall apply to all reactors irrespective of site and power.
* Land contamination, which impedes the use of large areas for a long period, is to be prevented.
* Deaths in acute radiation disease shall not occur.
* Incidents of extremely low probability are not to be considered.

The requirements may be considered as fulfilled if a release is mitigated to, as a maximum, 0.1% of core inventory of the cesium isotopes 134 and 137 contained in a reactor core of 1800 MW thermal power provided that the other nuclides of significance from the point of view of land contamination are retained to the same degree as cesium.
In order to meet the requirement an accident management strategy was defined aiming at a final stable state in which the core is cooled (preferably in the reactor pressure vessel but in the worst case on the containment floor) in an intact containment at low pressure.

In this paper this work is described for Ringhals 2, 3 and 4, all of them PWR, Westinghouse design, with a large dry containment. The measures implemented by 1988 are described in section 2, measures implemented since then are described in section 3 and finally remaining issues are discussed in section 4.

2. IMPLEMENTED MEASURES BY 1988

Measures implemented by 1988 was mainly a mitigation program, consisting of hardware measures, emergency operating procedures (EOP) and finally training of operating personnel.

2.1 Hardware measures

The plant modifications needed to fulfill the government requirements consist mainly of adding a redundant diesel driven water injection system operating through the containment sprays and a filtered containment vent. As a design basis case for the new systems total loss of all core cooling (including loss of all AC power) was chosen. In essence the accident management strategy means that when all attempts to terminate the accident and recover the core cooling in the pressure vessel have failed, water is injected into the containment until the bottom of the reactor vessel is covered by water and cooled. Containment venting is used to control the containment pressure and prevent overpressurization of the containment. The strategy is illustrated in Figure 1. A more detailed description is given in ref 1 and 2.

![Figure 1: Implemented Severe Accident Mitigating Measures at Ringhals PWR](image-url)
2.2 Emergency Operating Procedures (EOP)

At the time of the TMI accident Emergency Operating Procedures (EOP) were event oriented. As the events in the TMI had shown these EOP's needed improvement and within the Westinghouse Owner's Group new symptom based procedures were developed. These were called Emergency Response Guidelines (ERG) and was implemented at nuclear power plants of Westinghouse design, including Ringhals 2-4. The ERG's are relevant for the operators up to the time of vessel melt-through.

As the mitigating devices described in section 1 were developed, it became obvious that the EOP's must be further developed also to cover the time up to containment failure, as the key function of the new devices is to protect the containment. This observation led to the development of Beyond ERG (BERG), which will be initiated by a high temperature in the reactor core. See also ref. 3.

Below is the structure of the BERG's given, a general outline of the procedures is given, and the interface to the Site Emergency Plan.

Structure of the BERG

The main parameters to control in the severe accident procedures are the containment water level (to cool the melted core) and the containment pressure.

Therefore prior to reactor vessel failure, a sufficient water level must be established to allow quenching of the melted core, when it will fall into the reactor cavity.

When the reactor vessel has failed, a level allowing corium decay heat removal either by vaporization or by ECCS recirculation has to be maintained.

The normal containment spray system (injection mode) and the alternate spray system are possible systems to achieve the desired level depending upon their availability. However, operation of the alternate spray must be limited because it introduces pure water in the recirculation sump.

Obviously, one wants to limit the sump boron dilution to prevent the core from returning critical, should the recirculation system be eventually used to provide in-vessel core cooling. The constraint on alternate spray operation becomes irrelevant after reactor vessel failure because the corium no longer has a critical geometry (i.e. 100% pure water can be used to cool the core debris).

For these reasons, two "severe accident" procedure series were developed:

* The BERG-1 series which handles the short term period until reactor vessel failure has been diagnosed.

* The BERG-2 series which addresses the long term after reactor vessel failure until the inherent containment heat sinks will be able to match decay heat.

Each series is made up of two procedures. The selection between the two procedures of a series depends on the containment hydrogen concentration. Should the containment hydrogen concentration be less than 10% in dry air, a systematic containment depressurization by steam condensation is acceptable because the possible hydrogen burn that may be initiated will not induce a containment pressure spike of sufficient magnitude to actuate the containment vent system and will not result in creating additional radioactivity releases to the containment.
On the other hand, if the hydrogen concentration is in excess of 10% in dry air, a more cautious approach to containment pressure reduction is required, due to the possibility of a hydrogen burn which results in the automatic actuation of the containment vent, and Technical Support Center (TSC) evaluations will be required.

The general structure is shown in Figure 2.

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**Figure 2: Structure of ERG (Emergency Response Guidelines) and BERG (Beyond ERG)**

### 2.3 Emergency organization

The responsibility for the plant organization and actions in the case of a severe accident rests with the gen. An accident management system or organization is expected to work in a very complex situation where many different tasks have to be performed simultaneously. It is natural that the probability of successful termination of the accident will be higher if accident management schemes can be kept as logic and straightforward as possible.

Two basic demands have to be fulfilled during an accident:

- The demand from the affected unit for response to the accident and resolving the situation.
- The demand from authorities and the public for information guiding the off-site response.

The emergency organization should make clear the different roles with respect to these two basic demands. As a general rule the same people should not have the tasks of both supporting the affected unit and the authorities/public. One key to good accident management is to have the right people in the right place with appropriate tools and doing what they are trained for. In addition, the needs for communication and information transfer
within the organization must be satisfied in a well prepared and organized way so that the
fulfilment of the basic demands is not adversely affected.

The emergency organization of Ringhals have been formed having in mind the principles
discussed above. In order to meet the demand for effective response at the affected unit the
principle chosen is to deviate as little as possible from the normal organization. A good
organization will be characterized by people concentrating on what to do and how - not
who is going to do it.

The relevant parts of the Ringhals emergency organization are shown in Figure 3.

![Emergency Control Center Site Diagram]

**Figure 3: Onsite emergency organization at Ringhals PWR**

The unit manager is working in close contact with the plant operations manager in the
Emergency Control Center (ECC). Generally speaking the unit manager concentrates on
short term actions (within the next hours) in his unit, while the plant operations manager is
more concerned with the long-term strategy and site-related actions.

The information to the authorities and further to the public is the responsibility of the site
general manager and his information coordinator. The ECC is in close contact with the local
authorities responsible for emergency actions in the surrounding area and for contacts with
media. In the Ringhals organization it has been attempted to be able to supply proper and
correct information from the affected unit without interfering with the technical part of the
accident response.

### 2.4 Training requirements

A considerable effort was performed to train the emergency personnel in the basic facts of
the severe accident phenomenology and timing. The shift personnel were trained for the
BERG's and the emergency organization was trained for the site emergency plan. All categories were also educated for the radiological situation in the event of severe accident.

Today yearly retraining of the shift crew consists of:

* Repetition of implemented systems and Post Accident Sampling System (PASS)
* Technical specifications for relevant systems for accident management
* Simulator training for the BERG's. All parts of the ERG and BERG instructions are exercised within a six year cycle

All other people in the emergency organization have to participate in a severe accident management course every fourth year.

3. FURTHER MEASURES SINCE 1988

Since 1988 further work has been performed, in particular:

- Improved Core Damage Assessment (CDA)
- Implementation of a knowledge based handbook, called "Handbook for Unit Manager"
- Further analysis of potential threats to the containment, such as hydrogen combustion and Direct Containment Heating
- A PSA study, level 2 completed
- A long-term accident management analysis completed and implemented.

Each work is briefly described below.

3.1 Core Damage Assessment

Crucial decision points, if a severe accident would occur, are connected to water level in reactor vessel and vessel failure. One help is to determine the extent of core damage, so called "Core Damage Assessment", CDA. Several CDA-methods have been developed for all nuclear power plants in Vattenfall (Ringhals and Forsmark) such as:

a) Process parameters
b) Doserate in the containment
c) Activity content
d) Hydrogen concentration

Methods b) and c) are the main methods (method b) being quick but less accurate) while the other two are regarded as complementary. A more detailed description is given in ref 4.

3.2 Handbook for Unit Manager

In the later stage of an accident where plant funcions are severely degraded EOPs are no longer applicable and the accident management must primarily be knowledge based. The strategy is shown in Figure 4. The actions taken are guided by the knowledge of the emergency team and the main tool used is the Handbook for Unit Manager. In this handbook the most important results and conclusions from severe accident analyses performed for the plants are summarized under different headlines with references to background documents. It also contains strategies for handling various severe accident situations and points out important factors with respect to accident management.

The handbook is described in more detail in ref 5.
3.3 Potential early threats to the containment integrity

In a PWR the most important phenomenological early threats to the containment are hydrogen deflagration and Direct Containment Heating (DCH). Both these issues have been studied in more detail since 1988.

*Hydrogen deflagration*

Hydrogen is mainly generated by zircalloy oxidation of the fuel cladding during fuel meltdown but also during core-concrete interaction. As a PWR containment is filled with air during normal operation the hydrogen might burn, causing an over-pressurization of the containment. Several hydrogen deflagration calculations with the CONTAIN code have been performed and the results are summarized in ref 6. The conclusion is that hydrogen deflagration leading to containment failure has a very low probability, representing a residual risk. However, further studies will be performed as described in section 4.

*Direct Containment Heating (DCH)*

This phenomenon may occur in case vessel melt-through occurs with a high pressure in the primary system. The core melt might in the worst case become finely fragmented leading to direct heat-up of the containment atmosphere and oxidation of the metal in the melt, leading to rapid overpressurization of the containment. Calculations have been performed with the CONTAIN code (ref 7 and 8) leading to the conclusion that extreme assumptions had to be made to fail the containment integrity by DCH. Therefore this is regarded as a residual risk.

3.4 PSA, level 2 study for Ringhals 2

A PSA, level 2 study, has been completed for Ringhals 2, giving further insight in severe accident management (ref 9). One important issue in the study is the question of the melt coolability in the cavity below the vessel.

3.5 Long-term Accident Management

The aspects of accident management discussed so far mainly concern the first hours and days into an accident. It is important, however, when developing and implementing an
accident management program, to keep in mind also the long-term perspective. Otherwise the short-term actions may cause unnecessary problems and irreparable obstacles for the long-term handling of the plant.

In Sweden the long-term aspects of accident management issues have been studied in a separate project completed in 1991 (ref 10). The study was carried out for Ringhals 3/4 (and a BWR) postulating total loss of all core cooling, including loss of all a.c. power as the initiating event of a core melt sequence. The study comprised the time up to 5 years after the initiation of the accident.

Important conclusions were:

A) The final water level should reach the bottom of the reactor vessel but not more than 5 m above this level to ensure operational conditions for containment spray pumps.

B) The temperature in the containment should be less than 100 °C to minimize corrosion.

C) The pH should be between 10 and 10.5 to minimize corrosion and minimize iodine revolatilization.

For further details, see ref 10.

4. WHAT ARE THE REMAINING ISSUES?

Besides a continuous evaluation of existing EOP’s the most important issues remaining are:

- Implementation of hydrogen mitigation measures
- Core melt coolability in the cavity
- Water level in the reactor pressure vessel measurement
- Vessel failure criteria

These issues are treated below.

4.1 Implementation of hydrogen mitigation measures

As mentioned in section 3.3, hydrogen deflagration leading to loss of containment integrity is regarded as a residual risk. However, because of the technical development in this area, especially in Germany and Canada (ref 11), this issue is reconsidered. Today it seems that this issue is possible to eliminate for a reasonable cost. Therefore a site specific investigation will be carried out for Ringhals PWR to understand possible methods (igniters, passive recombiners or venting), feasibility and costs involved to implement hydrogen mitigation.

4.2 Core melt coolability in the cavity

This issue has been one of the dominant severe accident issues both for BWR and PWR in Sweden and has been important to verify. The Swedish strategy has been to fill water to the cavity region before vessel melt-through. As can be seen in Figure 5 there is a door blocking the passage of water from the containment to the cavity region. A flap has therefore been installed in this door to let the water enter the cavity region. However, it has not been verified in a total black-out case that water will reach the cavity region as no safety injection has pumped in more water to the containment in this case. Condensed steam from primary water leaving through the quench tank is probably not enough to fill the cavity with water.
Figure 3: Necessary water level to flood the cavity

In this context it was judged that water in the cavity, resulting in a steam explosion in the cavity, which could impair the containment, was extremely remote. However, the coolability issue has been the subject of considerable international controversy and therefore Vattenfall (jointly with other utilities and the Nuclear Power Inspectorate in Sweden in the so called APRI project) participates in the ACE, ACEX and CSARP projects. In particular, results in the experiments in the ACE project (unfortunately delayed) have been eagerly expected.

In case the ACE experiments give negative results concerning coolability in Ringhals PWR certain measures have to be investigated. For example, in the level 2 study for Ringhals 2, it was assumed a higher probability of cooling the melted core if water had entered the cavity before vessel melt-through. If further study indicates that this is correct a fairly easy improvement would be to have the cavity filled with some water during normal operation. Another possibility would be to investigate the possibilities to keep the melted core in the vessel, either by enhancing the possibilities to fill the vessel with water or by cooling the vessel from outside the vessel, if possible.

4.3 Reactor vessel water level

As we know from the TMI accident it might be essential for the operator to know the water level in the primary system. As has been mentioned before the most important information of the core status is obtained from the Core Exit Thermocouples. Another possibility in Ringhals 2, where gamma thermometers have been installed, is to obtain indications of the water level from these instruments (although they are not qualified for this task). The most important work in Sweden has been to develop a core cooling monitor, called BCCM (Becker's Core Cooling Monitor) for BWR (ref 12). It has not been investigated if these are possible to implement in a modified form in a PWR.

This issue may need further investigation.
LITHUANIAN VIEWS AND ACTIVITIES IN
THE AREA OF SEVERE ACCIDENT
MANAGEMENT IMPLEMENTATION

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NUCLEAR INFRASTRUCTURE IN LITHUANIA

Lithuania had taken over or newly set up organizational structures for the following functions:

* Operation (old + new)
* Administration (old + new)
* Regulation (new)
* Monitoring (old + new)

In addition, an organization had been established to provide technical consultation to all of the above structures:

* Technical support organization (new) - Ignalina Safety Analysis Group (ISAG)
IGNALINA SAFETY ANALYSIS GROUP

The ISAG was established in the Lithuanian Energy Institute in March 1992. It is the first group of professionals devoting their full time to safety aspects of the Ignalina NPP. Presently there are 12 thermal - hydraulic and heat transfer as well as material behavior specialists (9 doctorate level) and 2 support personnel.

The ISAG has the separate budget line.
ISAG ACTIVITIES

* Analysis of loss-of-coolant accidents and operational transients, using ATHLET and RELAP5 codes

* Assessment of the Accident Confinement System using DRASYS and CONTAIN codes

* Structural responsibility of the Accident Confinement System, using ALGOR codes

* Development of the RBMK-1500 neutron dynamic models and analysis of reactivity related accidents at different operating power levels of the reactor, using QUABOX/CUBBOX-HYCA code

* Probabilistic safety assessment of Ignalina NPP, using IRRAS and RISK SPECTRUM codes
Figure 1. Schematic representation of one loop of the main circulation circuit
1-separator drum, 2-downcomers, 3-suction header, 4-suction piping of the MCP, 5-MCP tanks, 6-pressure piping of the MCP, 7-bypass between headers, 8-pressure header, 9-group distribution header with flow limiter, check valve and mixer, 10-water piping, 11-channel to core, 12-fuel channel, 13-channel above core, 14-steam-water pipes, 15-steam pipelines
(all dimensions in meters)
ATHLET IGNALINA NUCLEAR POWER PLANT MODEL

* An ATHLET model of the Ignalina NPP for analysis of thermal-hydraulic transients as well as for accident analysis has been developed.

* Benchmarking analysis of natural circulation event was conducted. Calculation performed with this model compare favorably with plant data.

* Further model verification as well as development of the reactor control system and plant safety system is required.
CONCLUSIONS

* ATHLET and RELAP5 models of the Ignalina NPP have been developed. The models include the reactor primary coolant system, reactor control and plant safety systems required for the analysis of accidents as well as for operational transients.

* Benchmarking analysis of loss of all operating circulation pumps event were conducted. Calculation performed both with ATHLET and RELAP5 models compare favorably with plant data.

* Analysis of the natural circulation phenomena in the main circulation circuit and analysis of main safety valves LOCA for current plant situation were performed. In the all investigated cases the natural circulation conditions with reliable core cooling were realized.

* Further model verification as well as development are required to cover reactor transients.
ACTIVITIES AND PLANS IN FIELD OF SEVERE ACCIDENTS

* Became a member of Cooperative Severe Accident Research Program (CSARP)

* The US NRC severe accidents codes SCADAP/RELAP5 and CONTAIN have been acquired and are being installed

* ISAG personnel training in the operation of the US NRC codes MELCOR, CONTAIN and SCADAP/RELAP5 are performed

* CONTAIN model of the Accident Confinement System of the Ignalina NPP was developed. The short term future activities
  - improving the CONTAIN code capabilities
  - analysis of potential fission products transport in the Ignalina ACS using CONTAIN and SCADAP/RELAP5 codes
Implementation of severe accident measures in
The Netherlands.

George Vayssier,
Nuclear Safety Department,
The Netherlands.

OECD Specialist Meeting on the Implementation of
Severe Accident Measures.
Niantic, Conn., USA
June 12-14, 1995
Consideration of severe accidents measures in the Netherlands started in 1984, when it was planned to build 2000 MWe additional nuclear power. A guideline was developed recommending measures to reduce what is called 'the residual risk', i.e. the risk associated with severe accidents. To develop such a guideline was, in itself, a consequence of the TMI accident in 1979.

After the Chernobyl accident the expansion programme was halted and, instead, a reassessment of the safety of the existing nuclear power stations was performed. The assessment, which was made by GRS, Germany, concluded:

- it was prudent to install several severe accident mitigation features;
- it was strongly recommended to further strengthen the capability to cope with design basis accidents and to add flexibility for accidents beyond the design basis.

As a consequence, a large programma was started with the following objectives:

- reassess the design basis and the plant capability to cope with design basis events;
- reassess existing and eventual new systems so as to add flexibility in coping with accidents, both inside and outside the design basis;
- add features to mitigate severe accidents.

The programmes of both nuclear power stations are well under way. They are scheduled to be completed by 1997. Costs are for the Dodeward BWR US $ 60 million and for the Borssele PWR US $ 300 million, at current exchange rates.

Here it should be noted that the Netherlands licensing system is based on amended IAEA Safety Codes and Guides. The Code on Design mentions, in its revision 1, severe accidents in the sense that it is prudent to consider these events for the design of nuclear power plants. The licence of both power stations explicitly requires these Codes and Guides to be followed to the extent practical, so there is also a legal basis to require the consideration of severe accidents.

In a letter to both utilities in December 1988, a number of measures was required explicitly. Briefly, these were:

- a primary depressurization system capable of preventing high pressure meltthrough (HPME) in the event of a core melt; this system should have the dual function of also making bleed and feed possible;
- a filtered vented containment system; this system should also be able to depressurize the containment before a foundation meltthrough is to be expected; further it is required that this system should be compatible with the hydrogen mitigation system to be selected (see next item);
- a system to prevent or to cope with hydrogen combustion;
- adequate procedures to carry out the accident management measures.

These requirements will be met by the following features:

1. Dodewaard BWR.

1.1 Primary system depressurization will be done by two additional motor operated valves to prevent HPME. In addition, procedures guide the operators to early depressurize the primary system by the usual safety relief valves, in order to have low pressure feed easily available.

1.2 A dedicated containment filter was considered not to be necessary. Dodewaard uses, instead, a combination of a drywell spray system that operates using a separately powered fire extinguishing system, and a hardened vent system, equipped with standard ventilation filters (charcoal).

1.3 The containment was already in the early 1980s fully inerted, so the hydrogen problem is not acute; for the long term a thermal hydrogen recombiner is available; the containment filter will be able to cope with hydrogen.

1.4 Procedures designed to cope with severe accidents have not yet been written; symptom based procedures are already in place.

2. Borssele PWR.

2.1 Primary system depressurization will be achieved by a complete new set of primary safety and relief valves, using tandem safety valves of French design, which are also capable of discharging water and two-phase flow.

2.2 A dedicated containment filter will be used. It consists of a wet scrubber plus a dry fine filter.

2.3 Catalytic hydrogen recombiners will be installed. These do not fully prevent combustible concentrations, hence additional measures may be necessary. At present three options are studied: deliberate ignition using spark type igniters, post-accident
inertisation using carbon dioxide and 'early venting', using the containment filter. This last strategy reduces the containment pressure early in the event to keep eventual pressure rise during combustion below acceptable limits. Basically, it requires only an effective containment filter with a large capacity, since depressurisation actions may start already several hours into the accident.

The post-accident inertisation may be limited in magnitude, since already dilution of the containment atmosphere with inert gases reduces the combustion loads substantially, as was demonstrated in experiments, carried out by TNO Applied Research in the Netherlands on our request.

2.4 Severe accident procedures have not yet been written; symptom based procedures are in place.

Work still to be done.

Most work is in the detailed design phase. Only the strategy of the hydrogen mitigation is not yet complete. To solve this problem, expert solicitation took place on the different options. A comparison was made of different techniques with respect to:
- expected efficiency;
- research and engineering work still necessary to complete the different options;
- compatibility with the existing installation;
- costs.

The expert opinions are right now weighted in a computerized mathematical decision process, which will finally yield a 'best' option for Borssele.

All three options are basically acceptable for the regulatory agency. If the necessary research work for an option is not yet complete, the Borssele licensee will be allowed to select one now, provided that they commit themselves to obtain access to the remaining research and engineering work and transform it to their plant. For deliberate ignition a major item here is the composition of the gas mixture and the nature of the combustion process in the hydrogen source compartment of the containment.

The A/M procedures are still to be developed.

A last action that may be worthwhile to mention is that we try to set up a method to predict of what actually is going to happen in a severe accident, to support the emergency preparedness organisation. To this end use will be made of the PSAs that exist for both our power plants. The A/M methods described here and the associated operator procedures will be fed in into this work in the future.
KRŠKO NPP INDIVIDUAL PLANT EXAMINATION (IPE) INSIGHTS AS A KNOWLEDGE SOURCE FOR ACCIDENT MANAGEMENT DEVELOPMENT

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ABSTRACT:

Insights which could be screened out from PSA analysis form a good basis for further accident management development. The paper briefly presents and discusses the main results of the Krško NPP PSA Level 1 and 2, which deserves attention of all those who are concerned with risk and accident management.

INTRODUCTION

The Krško NPP is the only nuclear power plant in Slovenia. It is a two-loop Westinghouse design from the seventies. It was first connected to the grid in 1981. During almost 14 years of operation many provisions have been implemented which can prevent, mitigate or at least moderate the risk of severe accidents. These include improvements based on a TMI Action Plan, improvements in the fire protection program, installing of seismic instrumentation and high range monitoring of containment (gamma dose rate, pressure, hydrogen), post accident sampling, N16 monitoring in each steamline, primary Bleed and Feed, main control room monitoring for toxic gases, and installing a new plant process computer. In addition, the plant vulnerability by air attack has been studied. The Krško NPP follows WOG based Standard Technical Specifications and has already implemented the following procedures: Abnormal Operating Procedures (AOP), Emergency Operating Procedures (EOP) and Radiological Emergency Response Plan - Emergency Implementation Procedures (EIP).

In response to a SNSA decision the NPP Krško carries out PSA analysis as it is defined in GL 88-20. The first revision of Level 1 - (internal events) was submitted at the beginning of 1994, Level 2 at the end of 1994, and the shutdown modes in the second half of 1994. The system success criteria and severe accident scenarios analysis was performed by MAAP-3B code. In the frame of PSA external events analysis, a new seismic investigation of the area has been made and new seismic hazard curves have been determined.

The Krško NPP is currently working on a revision of IPE Level 2 (based on changes in Level 1 due to IAEA-IPERS mission comments and suggestions). The inputs for the plant-specific Severe Accident Management Guidance (SAMG) are the generic WOG SAMG, and the NEK-IPE Insights. The IPE insights will be extracted by the end of this year. The Krško NPP will start plant-specific SAMG development by the beginning of 1996. Consequently, some changes of EOP are expected. It has to be mentioned that Slovenia has its own regulations in the field of radiation protection and nuclear safety, which include also some probabilistic safety goals, for example, radiation risk for individuals outside the plant location is prescribed to be less than 10 microsievert per year. Radiation risk is defined as the product of accident probability and the consequences expressed in the dose to the most exposed member of the public. On the other hand the USA regulations are extensively used as a complementary standard.

In the meantime, the IPERS mission performed a snapshot review of Level 1. In the frame of EC PHARE - RAMG program, experts from Italian and Belgian regulatory bodies supported the
reviewing process of PSA at SNSA by a methodology transfer. The main objectives which should be achieved through the PSA review could be stated as follows:

- to be familiar with all potentially important details, such as, assumptions, modelling related to different areas and levels, data treatment, quantification, uncertainty and sensitivity,
- to gain confidence in the work that has been done and implicitly in PSA results; this is the most important objective, especially in view of future PSA applications, and the safety issues which will be drawn from them,
- to achieve a deeper insight into safety concerning the Krško NPP and to identify potential safety issues.

The information obtained from the results of PSA and severe accident studies has to be systematically examined and extensively used for developing an accident management strategy.

THE NPP KRŠKO PSA ACTIVITIES TIMEFRAME:

1986

Beginning of PSA (selected systems)
Coordination: NPP Krško under sponsorship of IAEA
Participants: Jožef Stefan Institute, Ljubljana
Rudjer Bošković Institute, Zagreb
Faculty of Electrical Engineering, Zagreb

June 1991

SNSA issues a PSA Decision with the following requirements:

<table>
<thead>
<tr>
<th>Requirement</th>
<th>Due Date</th>
</tr>
</thead>
<tbody>
<tr>
<td>PSA Level I</td>
<td></td>
</tr>
<tr>
<td>- internal events (power op. mode)</td>
<td>1 Sept. 1993</td>
</tr>
<tr>
<td>- internal events (all other modes)</td>
<td>1 June 1994</td>
</tr>
<tr>
<td>- external events</td>
<td>31 Dec. 1994</td>
</tr>
<tr>
<td>PSA Level II</td>
<td>1 Oct. 1994</td>
</tr>
<tr>
<td>Revision of SAR</td>
<td>end of each phase</td>
</tr>
<tr>
<td>&quot;Living&quot; PSA</td>
<td>end of each phase</td>
</tr>
</tbody>
</table>

1991

NPP Krško International Bid Invitation

1992

Selection of Contractor

Contractor: Westinghouse El. Corp.
Subcontractors: Jožef Stefan Institute, Ljubljana
Rudjer Bošković Institute, Zagreb
Faculty of Electrical Engineering, Zagreb
INETEC, Zagreb
Institute for Structural and Earthquake Engineering, Ljubljana
EQE, USA

Contractor for shutdown modes PSA (ORAM methodology) was ERiN, Engineering and Research, Inc.

All these requirements have been fulfilled except for external events analysis which will be concluded in the near future.
SOME NEK-IPE LEVEL-1 INSIGHTS CONSIDERATIONS:

The Core Damage frequency (CDF) is not the most important result of the PSA. Nevertheless, the estimated CDF for the NPP Královo is 5.047 E-5. Some other, more refined and specific results are more useful in terms of plant safety concern identification, backfitting, accident management measures allocation, and other applications for operational safety improvements. The contribution of the initiators is: 1/3 contribution to CDF is due to LOCAs, 1/3 to Loss of Offsite Power (LSP) and Station Blackout (SBO), and 1/3 to other initiators. The sequences of four IE (Medium LOCA, SBO, LSP, and SGTR) contribute to CDF more than 60%. Plant specific IE categories which induce high contribution to CDF are: loss of offsite power/station blackout, loss of component cooling and transients with main feedwater available. Many of the high ranked Initiating Event (IE) categories require intensive operator actions during accident coping. Apparently, many of the dominant cutsets and sequences contain in some cases up to three human action failure events (mainly recovery actions). This required greater investigation effort directed towards specific human action failure sensitivity study, review of related procedures and practice in the plant. Examples of such events are: Offsite power restoration, ECCS switchover to recirculation mode, CCW restoration, RCS cooldown and depressurization failure.

The hardware failures which are involved in cutsets with high impact to CDF are the plugging of the Containment Sump, Turbine Driven Pump failure, Diesel Generator failure, and Main Steam to Auxiliary Feedwater Pump isolation Valve. The most dominant sequences belong to the same IE as noted above. The most frequent elements of such sequences are: High Head Safety Injection System (injection and recirculation phase) failure, Auxiliary Feedwater System failure, Diesel Generator (DG) failure, Feed & Bleed, different operator actions (including recovery actions). If we take into account the part of sensitivity study and the importance measures, we can conclude that the Human Errors related to the establishment of the Auxiliary Feedwater System are highly contributive for maintaining the achieved CDF. Definitely, the main issues which occur are the SBO/LSP IE and related initiating events, such as DG and AFW systems failure and related human actions. If we look at the SBO/LSP safety concern more closely, the risk measures show that the DG reliability is very important and has to be at least maintained at the existing level. In addition, improvement of network and DG reliability greatly contributes to CDF reduction.

SOME NEK-IPE LEVEL-2 INSIGHTS:

Level 2 PSA provides plant specific means of investigating quantitatively the impact of various accident management measures. To illustrate the use of the PSA in this way, three sensitivity cases are presented, which investigate, in addition to the base case, the impact of various proposed accident management measures on the Královo NPP risk profile. Case 1 investigates the impact of allowing water to flood the reactor cavity at low pressure (non-dispersive) sequences. Case 2 investigates the effect of installing a filtered vent system, which is assumed to operate with 100% reliability, the purpose, of which is to prevent containment failure for late overpressure accidents. Case 3 investigates the effects of implementing both of the above. The wet cavity option reduces significantly the percentage of the basemat penetration (from 16.44% to 4.6%) and increases the number of no containment failures (from 21% to 40.41%). Thus the "small release" frequency decreases to 24.1% and the "very small release" frequency increases to 45.4% with no change in large release frequency. If the filtered vent system is implemented, then the very small, small and large release fractions become 58.8%, 10.6% and 30.5%, instead of 38%, 32% and 30.5% in the base case.
The third sensitivity case shows the very small release fraction is now 66.3%, the small release fraction 3.2%, and the large release fraction remains at 30.5%. Note that none of the cases presented affects the containment isolation failure/bypass release contribution.

Main Findings of PSA Level-2:

1. Nearly 70% of the core damage events lead to a fission product release which can be described as small. This result could demonstrate the robustness of the Krško plant and its resistance to severe accident challenges.

2. Very small releases result from core damaging events where the containment is not damaged (release categories RC1 and RC2) and where concrete attack occurs leading to very long term (many days) basemat failure (release category RC4). These events make up 38% of the core damage frequency.

3. Small releases result from late containment overpressure failure events (release categories RC3A, RC3B, RC5A and RC5B), which make up further 32% of the core damage frequency.

4. 30.5% of core damage frequency are "large releases". These include early containment failures (RC6), containment isolation failures (RC7A and RC7B) and containment bypass events (RC8A and RC8B).

5. Early containment failure (RC6) can be attributed as negligible 0.02% of CDF. This demonstrates that the Krško containment is very resistant to early failure challenges (including those due to high pressure melt ejection) and that the frequency of such challenges is low.

6. Containment isolation failure events (RC7A, RC7B) with 11.2% and bypass events (RC8A, RC8B) with 19.3% of CDF contribute significantly to the large release frequency, and are important contributors to total risk.

7. Dry reactor cavity design influences significantly the risk profile, leading to a relatively large fraction of long term concrete attack and basemat penetration sequences. Sensitivity analysis shows that a change to a wet cavity could decrease basemat penetration sequences from 16.4% to 4.6% of core damage frequency, and increase the "no containment failure" frequency correspondingly.

8. Formalized guidance for recovery actions after the occurrence of core damage (for example in the form of Westinghouse Owners Group Severe Accident Management Guidelines,) is not credited in the study since it is currently not implemented. Implementation of such guidance could bring significant benefit in reducing the overall risk. For example, strategies addressing the minimization of fission product releases, recovery of containment isolation, and recovery of key equipment could significantly reduce the frequency and consequences of the largest part of release categories RC3 to RC8.

9. The base case results indicate that the frequency of recovery of the damaged core in vessel is very small (0.4 of CDF). However, as noted above, the study does not credit accident management strategies, which are currently not being implemented. A strategy to flood the reactor vessel in reactor cavity and thereby prevent vessel failure could lead to a significant risk benefit. Such a strategy, implemented at other plants, is addressed
in the WOG SAMG and will be investigated as part of the SAMG implementation.

10. The impact of a typical severe accident filtered vent system in terms of its effect on the frequency of containment failure, is investigated in sensitivity studies. The system basically converts small release events (late overpressure failures) to very small releases (no containment failure, filtered release). The greatest benefit can be obtained from the system if it is used in combination with a wet cavity modification.

REGULATORY POSITION TO THE IPE APPLICATION

The core damage frequency of the Črčko NPP is comparable to CDF of other similar plants. Nevertheless, lower CDF is desirable. It is therefore necessary to utilize all effective ways to improve the plant operation safety and enhance the prevention and mitigation of severe accidents, and consequently to decrease the risk.

The most important specific actions to be taken are:

- to require from the licensee to decrease the CDF by measures which will decrease the level of the main contributors, primarily to develop measures against dominant accident sequences which have been identified,
- to require from the licensee to improve PSA models and the PSA data base, so as to be more accurately tuned with the plant specific design, SSC state, and operating practice,
- to extend the use of operational experience feedback in safety improvement,
- to use extensively the PSA in safety improvement, modification prioritization, and decision making process,
- to develop a system for risk based inspection,
- to develop a severe accident strategy which will be based on technological knowledge, and which will be effective on both, prevention and mitigation,
- to decrease uncertainties in the modelling of Črčko NPP severe accidents phenomenology by encouraging domestic research organizations to increase their efforts in developing stronger capabilities in severe accident research and to take part in international benchmarks,
- to require from the NPP Črčko to develop and implement plant specific accident management (SAMG) procedures,
- to avoid that accident management measures could deteriorate the existing safety functions or degrade the prevention level to the design basis events,
- to update to perform regularly, at least during the periodic safety review, the implemented accident management measures.

SNSA foresees five main short-term and long-term sets of actions which should be initialized to decrease a risk level due to plant operation and enhance the relations between the authority and the plant:

A) Change the risk profile (decrease a risk level) through the backfittting based on a wide application of the results of the PSA study and open-issues centred studies, such as Station Blackout study, Fire Hazard Analysis, etc.
B) Change the risk profile by the implementation of severe accident centred measures,
C) Increase the knowledge and research capabilities in severe accident understanding and modelling,
D) Incorporate the Probabilistic Safety Criteria/Goals into the new legislation which is under development,
E) Introduce a Periodic Safety Review into the new legislation and put it into practice.
1. Probabilistic Safety Assessment of Nuclear Power Plant Krško, (Level 1 and Level 2).
NEK PSA Level 1 Results

Percent Contribution per IE Category

Total Core Melt Frequency = 5.047E-5 / RxYr

Core Melt Frequency = 5.0E-5 / RxYr
NEK IPE Level 2 Results

Krško Release Category Definitions and Results
(results are shown as percentage of total core damage frequency in each release category)

<table>
<thead>
<tr>
<th>RC no.</th>
<th>Release Category Definition</th>
<th>Base case (plant as-is)</th>
<th>Sensitivity case 1 (wet cavity)</th>
<th>Sensitivity case 2 (blurred view)</th>
<th>Sensitivity case 3 (both)</th>
</tr>
</thead>
<tbody>
<tr>
<td>1</td>
<td>Core recovered in-vessel, no containment failure</td>
<td>0.39</td>
<td>0.39</td>
<td>0.39</td>
<td>0.39</td>
</tr>
<tr>
<td>2</td>
<td>No containment failure</td>
<td>21.11</td>
<td>40.48</td>
<td>21.11</td>
<td>40.41</td>
</tr>
<tr>
<td>3A</td>
<td>Late (time frame IV) containment failure, no molten core-concrete attack</td>
<td>0.60</td>
<td>0.64</td>
<td>0.0</td>
<td>0.0</td>
</tr>
<tr>
<td>3B</td>
<td>Late (time frame IV) containment failure, molten core-concrete attack</td>
<td>14.32</td>
<td>7.55</td>
<td>8.62</td>
<td>2.55</td>
</tr>
<tr>
<td>4</td>
<td>Basemat penetration (no overpressure failure)</td>
<td>16.44</td>
<td>4.56</td>
<td>16.44</td>
<td>4.60</td>
</tr>
<tr>
<td>5A</td>
<td>Intermediate (time frame III) containment failure, no molten core-concrete attack</td>
<td>14.63</td>
<td>15.48</td>
<td>0.0</td>
<td>0.0</td>
</tr>
<tr>
<td>5B</td>
<td>Intermediate (time frame III) containment failure, molten core-concrete attack</td>
<td>2.01</td>
<td>0.60</td>
<td>2.01</td>
<td>0.61</td>
</tr>
<tr>
<td>6</td>
<td>Early (time frame I or II) containment failure</td>
<td>0.02</td>
<td>0.02</td>
<td>0.02</td>
<td>0.02</td>
</tr>
<tr>
<td>7A</td>
<td>Isolation failure, no molten core-concrete attack</td>
<td>5.84</td>
<td>9.00</td>
<td>5.84</td>
<td>8.99</td>
</tr>
<tr>
<td>7B</td>
<td>Isolation failure, molten core-concrete attack</td>
<td>5.38</td>
<td>2.21</td>
<td>5.38</td>
<td>2.22</td>
</tr>
<tr>
<td>8A</td>
<td>Bypass, scrubbed</td>
<td>12.74</td>
<td>12.74</td>
<td>12.74</td>
<td>12.74</td>
</tr>
<tr>
<td>8B</td>
<td>Bypass, unscrubbed</td>
<td>6.51</td>
<td>6.51</td>
<td>6.51</td>
<td>6.51</td>
</tr>
</tbody>
</table>

Figure 5-4 - Release Categories for Base Case