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Use of PSA Level 2 analysis for improving containment performance

Report of a Technical Committee meeting held in Vienna, 9–13 December 1996

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FOREWORD

Containment performance could have an important role in determining the consequences of severe accidents in nuclear facilities and so containment capabilities are increasingly being analysed in probabilistic safety assessments (PSAs). These studies provide a basis for studying the effects of upgrading measures or of additional, new equipment intended to improve containment response and to reduce accident consequences. They are also used to investigate severe accident management.

In 1992 a Safety Series report on Probabilistic Safety Assessment was produced by the International Nuclear Safety Advisory Group (INSAG) as 75-INSAG-6. This report states that the methods and tools available to analyse containment response to severe accidents still contain significant uncertainties which make application of such studies difficult.

In order to discuss and exchange experience on different aspects of methods associated with Level 2 PSA and its applications for improving containment performance, the IAEA held a Technical Committee meeting in Vienna in December 1996. The meeting, which was attended by 26 participants from 20 Member States, provided a broad forum for discussion. The meeting addressed the issues related to the actual performance of Level 2 PSA studies as well as the insights gained from applications to improve containment performance. Particular attention was given to studies and applications for WWER type reactors, for which Level 2 work is still in its early stages, and for channel type reactors where modelling of accident progression is complex and significantly different from vessel type light water reactors.

This TECDOC contains the papers presented at the meeting and the results of extensive discussions which were held in specific working groups.

EDITORIAL NOTE

CONTENTS

1. INTRODUCTION

Level 1 probabilistic safety assessments (PSAs) have been or are currently being performed for most nuclear power plants (NPPs) worldwide. Level 1 PSA determines whether accident sequences may result in fuel or core damage. In Level 2 PSA the course of core damage sequences is further considered in terms of accident progression, physical phenomena within the containment, containment response and releases, as well as transport of radioactive materials within the containment and to the environment. The main task of the Level 2 PSA is to assess containment failure modes and source terms of radioactive materials to the environment of the plant. Source terms are given by the amount of different types of radioactive materials released and the characteristics of the release, e.g. release height, energy and timing. Based on the source terms, off-site consequences are then estimated in the Level 3 PSA.

In addition to the Level 1 PSAs Level 2 analyses have been already carried out, are in process or planned for all major reactor types. The information and results of these studies are used to identify weak points or vulnerabilities of the containment and containment systems under severe accident conditions. Level 2 PSA is used to study the effects of upgrading measures or of additional, new equipment intended to improve containment and containment systems response and to reduce accident consequences. It is also used to investigate severe accident management. Severe accident management tries to limit or mitigate severe accidents and their consequences by refining emergency operation procedures using either existing, newly added, or improvised equipment and systems.

The investigation and implementation of upgrading measures intended to improve containment and containment systems behaviour under severe accident conditions and to reduce the consequences of severe accidents cannot be done based on Level 2 PSA alone due to the limited level of detail and the uncertainties in methods, models and data. Dedicated detailed analyses of specific phenomena with specialized computer codes, engineering type studies and sometimes even experiments are necessary for this purpose. However, PSA provides the necessary overall framework to consider, for example, negative and positive impacts of planned upgrading measures and also allows to carry out cost–benefit analyses.

2. OBJECTIVES OF THE MEETING

The purpose of the Technical Committee Meeting was to bring about a broad exchange of information in the area of using Level 2 PSA for improving containment performance. Furthermore, it provided the opportunity to identify present capabilities and limitations of Level 2 methodology, in particular regarding the different types of NPPs and the corresponding confinements or containments. The specific objectives of the meeting were to review the current approaches, experiences and practices in Member States in the use of Level 2 PSA for improving containment performance, to discuss pertinent issues and to make recommendations on areas and aspects which need further development.

To achieve the above objectives, the Technical Committee meeting consisted of presentations and discussions. Special attention was given to aspects of Level 2 analyses for WWER and channel type reactors. Participants from 13 Member States took the floor on nine occasions and presented an additional nine papers. The papers are reproduced in an annex to this report.

3. REVIEW OF EXPERIENCE

This section contains summaries of individual presentations. Table I lists the PSA requirements in the Member States represented at the meeting. The presentations were organized in three sessions according to the types of reactors considered:

(1) Vessel type LWRs (with the exception of WWERs which were considered in Session 2, see below);
(2) WWER type reactors;
(3) Channel type reactors.

3.1. SESSION 1, VESSEL TYPE LWRs (EXCEPT WWERs)

In the first presentation, *A. D'Eer, Belgium*, described recent Level 2 PSA work for NPPs in Belgium. The plants considered were Tihange Unit 1, a 3-loop PWR, and Doel Units 1 and 2 which are 2-loop PWRs. The following specific accident management features and measures were considered for the Doel 1 and 2 units:

- hydrogen catalytic recombiners
- reactor coolant system depressurization
- reactor cavity flooding
- external vessel cooling
- independent containment heat removal.

It is planned to carry out detailed modelling and evaluation of plant specific severe accident management guides which are currently being drawn up. The performance of source term calculations, to be used as a basis for planning emergency exercises, is also contemplated.

The presentation of *M. Kajimoto, Japan*, summarized Level 2 studies performed for reference BWR and PWR NPPs. The calculated results show low containment failure frequencies and illustrate the high safety level which has been achieved for both reactor types. Accident management is addressed in terms of the concept of defence in depth. For early containment failure, the dominant accident sequences are anticipated transients without scram (ATWS) for the reference BWR, and steam generator tube rupture (SGTR) for the reference PWR. The phenomena of steam explosion, pressure spike at reactor vessel failure and direct containment heating (DCH) are not contributing significantly regarding dominant sequences that lead to containment failure. This includes consideration of uncertainty bands. Late containment failures with loss of heat removal are dominant accident sequences which have a relatively high frequency and source terms.

The paper of *M. Sobajima, Japan*, describes a proposal for accident management optimization based on the study of accident sequence and source term analyses for a BWR. Specific aspects of the presentation were the following:

- Accident management measures are to be implemented in all LWRs by 2000 in accordance with the regulator's recommendation and the utility's PSA Level 1.5;

- Source terms were evaluated by JAERI in all BWR sequences with the THALES code in which loss of decay heat removal resulted in the largest release;

TABLE I. REQUIREMENTS FOR PSA IN MEMBER STATES

NPP Type	Country	PSA Level 1	PSA Level 2	PSA Level 3
LWRs	Belgium	Required[a]	Partly required [b]	– [c]
	Brazil	Required	Voluntary	– [c]
	Germany	Required	Studies ongoing	– [c]
	India	Suggested[a]	– [c]	– [c]
	Islamic Republic of Iran	Required	Required	– [c]
	Japan	Strongly suggested	Strongly suggested	– [c]
	Republic of Korea	Required	Suggested	– [c]
	Mexico	Required	Required	Desirable
	Netherlands	Required	Required	Required
	Pakistan	Suggested	Voluntary	Voluntary
	Spain	Required	Required	– [c]
	UK	Required	Required	Required
WWERs	Czech Republic	Voluntary	Voluntary	Voluntary
	Hungary	Required	planned	– [c]
	Russian Federation	Required	Desirable	– [c]
	Slovakia	Required	Voluntary	– [c]
PHWRs	Canada	Suggested	Voluntary	Voluntary
	India	Suggested	Voluntary	Voluntary
	Pakistan	Voluntary	Voluntary	Voluntary
	Romania	Required	Required	Required
	Republic of Korea	Required	Suggested	– [c]
RBMKs	Russian Federation	Required	Desirable	– [c]

[a] 'Required': Officially required in the regulations.
'Suggested': Not officially required in the regulations, but the regulatory organization may very likely request the licensee to perform a PSA

[b] Limited to containment performance analysis.

[c] No requirement or no information available.

- Identification of the priority and importance of accident management measures was made for the sequences with larger risk contributions;

- Consideration for optimizing emergency operation guides is essential for risk reduction.

The contribution of *See Darl Kim, Republic of Korea*, described a systematic sensitivity study performed to assess the effects of advanced safety features of the Korean standard NPP. The specific advanced safety features considered designed to mitigate severe accidents were the reactor cavity flooding system, hydrogen igniters, and the containment filtered venting system. It was shown that the reactor cavity flooding system and the igniters do not enhance containment performance in a significant way. In contrast, the containment filtered venting system appeared to be effective. This result is thought to be a consequence of the specific containment design, in particular because of the relatively high failure pressure.

The paper of *R. Lopez Morones, Mexico*, considers the use of a relatively simple parametric model to estimate containment loads following an ex-vessel steam spike. Specific aspects of his presentation were the following:

- The study was initiated because several PSAs identified containment loads accompanying reactor vessel failure as a major contributor to early containment failure.

- The description of the simple but physically sound parametric model which was adopted to estimate containment loads following a steam spike into the reactor cavity.

The contribution of *P.J.T. Bakker, Netherlands*, described the PSA work carried out for the Borssele and Dodewaard NPPs. A full scope PSA Level 3 analysis for all NPPs was required in the Netherlands. Both NPPs fulfilled this requirement for their ‘as-built’ status. The methods applied were based on NUREG-1150 and used the MELCOR, MAAP, WAVCO, EVNTRE and NUCAP+ codes. Modifications to the containment will be made based in part on the Level 2 PSA results. The most important modifications concern filtered vent systems, dry well bottom protection and hydrogen recombination systems. The update of the PSA to the new layout shows a clear improvement of the containment performance. The Borssele PSA is also used to study AM (accident management) measures. The detailed study showed that the important core damage frequencies out of the Level 1 PSA are not the most significant in terms of the environmental impact according to the Level 3 PSA. The containment bypass sequences, with a low frequency but high consequence, can be reduced with appropriate AM measures. These AM measures based on the PSA lead to a further improvement of the ‘containment’ performance.

The paper of *R. Otero, Spain*, presents the increasing scope of PSA/IPE (individual plant examination) analyses required to the Spanish NPP by the Spanish Regulatory Commission (CSN), and the state of Level 2 analyses in each plant. The strategy envisaged for the implementation of severe accident management in Spanish LWRs was also shown in his presentation. The main phenomena and key equipment for containment performance were identified for two specific Spanish plants, a PWR and a BWR, whose Level 2 studies are either under preparation or development.

The presentation of *C.H. Shepherd, United Kingdom*, considered the full scope Level 3 PSA carried out for the Sizewell B PWR. This included a level 2 PSA to determine the effectiveness of the containment in preventing an uncontrolled release of radioactivity to the environment following core melt. The small containment event tree approach with 20 nodes in 4 time frames was used to model the phenomena which could challenge the integrity of the containment. The phenomena addressed included fuel coolant interaction, hydrogen burn, detonation, high pressure melt ejection, direct containment heating, etc. The phenomena were modelled using MAAP supported by other codes and taking account of recent research on these topics. The failure of the containment due to overpressure was modelled using a finite element analysis. This predicted that gross failure would occur at 2.6 times the design pressure due to failure of the cylindrical walls and that enhanced leakage would occur at 2.4 times the design pressure due to the liner tearing at the equipment hatch and personnel airlock. This was confirmed by a tenth scale model test. The analyses addressed a number of accident management measures and concluded that there was a significant benefit from recovery of the emergency core cooling system (ECCS) and the reactor building spray system and the use of the reactor building spray system to put water into the containment. However, the analyses determined that there would be no significant benefit from the incorporation of a filtered containment venting system.

3.2. SESSION 2, WWER TYPE REACTORS

The presentation of *L. Kucera, Czech Republic*, examined the results and experiences made in the Level 2 PSA for the Temelin NPP. The Temelin NPP is still under construction, therefore the PSA study has been based on a number of conservative assumptions. For this reason, the PSA Level 1 and Level 2 gave somewhat pessimistic results, such as large early releases due to a high contribution of containment bypass and early containment failure sequences. When realistic data on the plant is available (such as detailed design data, success criteria based on deterministic calculations, emergency operating procedures (EOP), etc.) and consequently the Level 2 PSA gives more realistic results, it is expected that these results will be widely used in the preparation of severe accident management guidance as well as in the discussion on measures to improve plant and containment performance.

Recently a project was started during which a database of the most probable and/or severe accidents was created based on results of the Level 2 PSA. In the event of an accident, this database should assist the technical support centre in understanding the progress of the accident and determining the measures to be taken.

The presentation of *B. Kujal, Czech Republic*, analysed the work related to WWER-1000 containment behaviour and loads during severe accidents. The systematic examination of the phenomena and mechanisms which can result in the loss of containment function in the course of severe accidents was performed in the Nuclear Research Institute Plc. Rez (Czech Republic).

The following phenomena were identified to be the most relevant:

- long term overpressurization
- hydrogen burns
- steam explosions
- direct containment heating
- corium penetration through the hermetic boundary.

The MELCOR, CONTAIN and WECHSL codes were used for analyses. Input model and input data files were prepared under quality assurance. Extensive analysis (including sensitivity studies) of all the phenomena mentioned above resulted in the following conclusions:

- Direct containment heating and steam explosions do not seem to be a real danger for the WWER-1000 containment.

- Long term overpressurization, hydrogen detonations and in particular corium penetration through the bottom floor plate of the containment are the real threats for the containment function.

The results of the analysis were used for severe accident management proposals and will be utilized in the revised version of the Level 2 PSA study for the Temelin NPP.

Level 2 PSA related studies in Hungary are described in the paper of *G.L. Horvath, Hungary*. A Level 1 PSA has been performed for the PAKS Unit 3 in the framework of the AGNES project, which is an overall safety reassessment project for this plant. The current PSA topics being considered are shutdown PSA and human reliability. Limited Level 2 PSA studies were also performed for the AGNES project. The analytical tools available for the Level 2 work are STCP, CONTAIN, MELCOR, RELAP-SCDAP, and MAAP. Most of these codes are running at VEIKI. Good experience is available for STCP, CONTAIN and MELCOR, and experience is being gained using MAAP and RELAP-SCDAP. Still more experience with all the codes is needed.

The applicability of available analytical tools for containment ultimate failure pressure analysis for the WWER-440/213 containment is limited. Current active analytical issues are: accident management, quick accident prediction tools (SESAME), hydrogen control and bubbler condenser (large scale) qualification. Level 2 PSA may be initiated in a few years. The importance of Level 2 PSA is seen for accident management and in the fact that Level 2 PSA may result in dominant sequences different from those of Level 1. Level 2 PSA may be most important for authorities and the plant to emphasize also the last barriers. Once most of the Level 1 PSA issues are implemented, Level 2 PSA may be started.

The paper of *I. Kalinkin, Russian Federation*, describes severe accident related work for the Balakovo Unit 4. The Level 1 PSA for the Balakovo WWER-1000 Unit 4 has been carried out and the Level 1–Level 2 PSA interface is developed and represented by a set of interfacing event trees (using the Risk Spectrum PSA code). The resulting plant damage state (PDS) frequencies were used as input to the containment event tree (CET) developed within the Level 2 PSA. Consequences of the CET are grouped into radiological release categories (RCs). Allocation of RCs in the CET is based on the similarity of the sequence characteristics. Each RC may be quantified in terms of the potential fractions of core inventory that may be released to the environment and also by the characteristic of releases such as time to release, release duration and availability of warning time. Containment capability analysis (with the ABAQUS code) was carried out and shows a high level of containment ultimate pressure. The containment fragility curve was developed taking into account major randomness and uncertainties sources in containment characteristics. Analysis of containment leakage and of the performance of penetrations was also performed. As a result, the containment failure mode under internal pressure rise was defined as a global failure of the containment in the membrane area of the cylinder. As the data on severe accident analysis was only available for a large loss of coolant accident (LOCA), the quantification of containment event tree (CET) sequences was limited to one sequence. The nodal probabilities were introduced based on data from the analysis of the large LOCA, containment fragility curve and expert judgement. The most probable sequence is associated with low pressure failure of the reactor pressure vessel (RPV) and basemat meltthrough through instrumentation channels with further melting of the foundation slab and release to the environment (probability $<10^{-7}$).

The presentation of *E. Shubeiko, Russian Federation*, addressed accident progression analyses for the Novovoronezh NPP Units 3 and 4. These units are of the WWER-440, V-179 type. The code used for these calculations was MELCOR. The confinement of this plant has a limited capacity and is equipped with pressure relief flaps to limit the pressure rise during accidents. Pressure reduction capability in the confinement is provided by means of a confinement spray with heat exchangers to remove heat. Therefore, it appears that ECCS functions and confinement performance are closely connected. In order to account for this dependency, the integral accident progression code MELCOR was already used in the Level 1 accident sequence analysis. The code allows to model confinement performance, confinement systems, the behaviour of the reactor and reactor coolant system (RCS) in an integral way. Two specific LOCA sequences were discussed, with leak sizes of 32 mm and 60 mm. It was pointed out that further analyses with a thermohydraulic code like RELAP are needed to determine, for example, the safety system requirements more precisely.

3.3. SESSION 3, CHANNEL TYPE REACTORS

The paper of *R.N. Bhawal, India*, deals with the use of containment safety features to reduce source terms for an Indian PHWR. Level 2 PSA analysis is used for evaluating source terms to the atmosphere by judging and analysing containment performance. By improving the containment safety features, the release can be reduced, resulting in risk minimization. Indian pressurized heavy water reactors (IPHWRs) adopt double containment philosophy extended to all penetrations and other openings such as airlocks. In addition to the special engineered safety features (ESFs), such as reactor building emergency coolers, a secondary containment recirculation and purge system (SCRP), a primary containment filtration and pump back system (PCFPB) and a primary containment controlled discharge system (PCCD) have been incorporated to enhance the containment performance. In this paper the effectiveness of these different systems and their interactions in improving containment function have been analysed and discussed. It is seen that the double containment and the SCRP system alone can drastically reduce the release of radioactivity. It is also seen that some of the ESFs are effective when operating automatically, while others can be operated manually for better containment performance, depending on the accident situation. The operator can effectively use PCFPB and PCCD in conjunction with other inputs like status of other equipment, activity level in the reactor building atmosphere, weather condition etc. Operators can be provided with appropriate software (plant specific) for on-line estimation of containment status (Level 2 analysis tool) and take appropriate management strategies for effective risk reduction/emergency preparedness.

The analytical tools developed and adapted for PSA Level 2 studies on containment system performance for the Indian pressurized heavy water reactor are reviewed in the paper of *S.K. Haware, India*. Pressure buildup in a multicompartment containment after a postulated accident, the growth, transportation and removal of aerosols in the containment and issues related to the hydrogen behaviour in containment are complex processes of vital importance in deciding the source term. The necessity of well tested analytical tools to enable proper estimation of the source term and assessment of containment system performance is pointed out.

While the CONTRAN code for containment system thermal hydraulic transient analysis has extensively been validated and NAUA MOD5, which performs aerosol behaviour analysis, has been adapted, the HYRECAT code for analysis of hydrogen behaviour and the SPARC code for suppression pool aerosol scrubbing have undergone a limited degree of validation. With the use of data from in-house planned tests and other tests, further validation of these codes is at hand. Development of an integrated code for containment system performance analysis is in progress.

D. Serbanescu, Romania, focused in his contribution on the role of PSA in the Romanian regulatory framework. Deterministic and probabilistic approaches are both used for the Cernavoda NPP, which is of the CANDU 6 type. Historically, the probabilistic part consisted of reliability analyses and SDMs (Safety Design Matrices). Furthermore, PSA has been requested for this plant, which was restricted to Level 1 for Unit 1, but was extended to Levels 2 and 3 for Unit 2. Accident management measures and features have to be considered as well. The assessment of containment performance needs to include seismic capacities due to the relatively high seismic hazard at the site. The importance of containment tests and inspections to ascertain containment status throughout the operational life was pointed out, as well as careful evaluation of related events at the plant.

The paper of *I. Turcu, Romania*, first summarizes the status of PSA activities for the Cernavoda NPP. The extension of the PSA work to include Level 2 PSA is mentioned. Important characteristics of this reactor type for Level 2 PSA were outlined. Due to the specific layout of the CANDU reactor, the evolution of severe accidents is considerably different from vessel type LWRs. Accidents can be roughly categorized into three categories, 'severe accidents' which lead to the loss of core structural integrity, delayed loss of core structural integrity as a consequence of the loss of heat sinks, and fuel channel failures. This paper describes the current work for modelling accident progression in the core region and outlines the elements for the Level 2 PSA including definition of PDSs, probabilistic containment logic and source term calculation. It is pointed out that uncertainties have to be considered which are contained in the models to bridge knowledge gaps. For this purpose, sensitivity studies will be carried out for key modelling assumptions.

The paper of *R. Gubler, IAEA*, describes Level 2 activities for RBMK type reactors. Probabilistic safety analyses (PSAs) of the boiling water graphite moderated pressure tube reactors (RBMKs) have been developed only recently and are limited to Level 1. Activities at the IAEA were first motivated because of the difficulties to characterize core damage for RBMK reactors. Core damage probability is used in documents of the IAEA as a convenient single valued measure, for example for probabilistic safety criteria. The limited number of PSAs completed for the RBMK reactors have shown that several special features of these channel type reactors call for a reexamination of the characterization of core damage for these reactors. Furthermore, it has become increasingly evident that detailed deterministic analysis of design basis accidents (DBAs) and beyond design basis accidents reveal considerable insights into RBMK response to various accident conditions. These analyses can also help in better characterizing the outstanding phenomenological uncertainties, improved EOPs and AM strategies, including potential risk-beneficial accident mitigative backfits. The deterministic efforts should first be directed towards elucidating accident progression processes and phenomena, and second towards finding, qualifying and implementing procedures to minimize the risk of severe accident states. The IAEA PSA procedures were mainly developed bearing in mind vessel type LWRs, and would therefore require extensions to make them directly applicable to channel type reactors.

4. USE OF PSA LEVEL 2 ANALYSIS FOR IMPROVING CONTAINMENT PERFORMANCE

The following particular specific uses were identified in the meeting regarding Level 2 analysis and containment performance assessment:

Plant, operator environment

- Identification of major containment failure modes and corresponding releases of radioactivity;

- Identification of human actions important for safety, related to Level 2 PSA;
- Identification of particular important phenomena and processes;
- Identification of specific features, systems, components and equipment of the containment important for safety;
- To gain insights into the progression of severe accidents;
- Identification of most effective areas for improvement;
- Investigation of qualification, capacity, operability, and accessibility of containment related equipment and structures;
- Basis for upgrading or backfitting of I&C for diagnosis of accident conditions;
- Assessment and planning the use of existing or improvised systems, investigation of benefits of existing or new equipment (including 'hardening' or upgrading of existing equipment and systems);
- Identification of operator actions to use existing containment related engineered safety features in an efficient way;
- Development of severe accident management features, systems and procedures;
- Development of emergency operating procedures;
- Evaluation of technical specifications regarding features important in the Level 2 PSA;
- Prioritization of inspection, testing, surveillance and maintenance related to Level 2 PSA;
- Support for decision making for design modifications, upgrades and backfitting, including cost-benefit considerations;
- Operator training;
- Planning of emergency exercises;
- Communication means between utility and regulator.

In general, Level 2 information and results are used as a complement for deterministic analysis and design base considerations.

Regulatory environment

- Evaluation of licensing practices;
- Use of Level 2 PSA in the licensing framework ('risk informed' regulatory policy);
- Support for regulatory decision making (e.g. comparison with probabilistic safety criteria);

- Development of severe accident policies;

- Regulatory inspection prioritization;

- Development of regulatory requirements regarding Level 2 PSA and applications.

Level 2 PSA research and methodology

- Basis for prioritization of research activities;

- Gaining insights on particular phenomena and processes which are important for containment and containment performance;

- Identification of limitations and uncertainties in current PSA Level 2 methods, knowledge base and models.

Level 2 PSA also provides the necessary information in terms of releases of radioactive material to assess off-site consequences in the Level 3 PSA.

4.1. ACCIDENT MANAGEMENT MEASURES FOR LWRs

A list of current accident management features and measures was compiled by the meeting participants. The list below gives an overview of the measures which are under study or are applied to LWRs for improving containment performance and to reduce source terms. The measures are partly based on practical applications in Level 2 studies. The list is restricted to the information obtained from the participating countries and does not pretend to be complete.

Cooling of a partly failed core

- Use of alternative water sources
- Use of alternative power sources
- RCS depressurization
- Water injection into steam generators

Prevention of induced SGTR

- RCS depressurization
- Injection into steam generators
- Isolation of the failed steam generator

Prevention of vessel failure

- Reactor cavity flooding
- RCS depressurization

Control of hydrogen

- Hydrogen recombination
- Hydrogen igniters
- Post-accident inerting

Containment overpressure prevention

- Alternative containment heat removal
- Filtered venting system
- Natural circulation cooling with air conditioner system
- Manually activated spray
- Hardened vent from the suppression pool

Basemat meltthrough

- Cavity/pedestal flooding (before and after vessel failure)
- Core catchers (advanced LWRs)
- Plugging vertical channels which penetrate the reactor cavity floor (WWER-1000)
- Provide a pathway for the molten material to flow out of the reactor cavity (WWER-1000)

Source term reduction

- Filtered venting system
- Water coverage of the leak in the failed steam generator

4.2. USE OF LEVEL 2 PSA ANALYSIS FOR IMPROVING CONTAINMENT PERFORMANCE — SPECIFIC ASPECTS OF WWER TYPE REACTORS

The PSA status for WWER type reactors was compiled by participants during the meeting. The current status of PSA and severe accident management guidance for WWER type reactors is given in Tables II and III below.

TABLE II. PSA STATUS OF WWER-1000 REACTORS

Country/Plant	**Level 1 PSA**	**Level 2 PSA**	**Level 3 PSA**	**Severe AM Guidance**
Czech Republic				
Temelin	Performed	Performed	Not planned	Introductory study
Russian Federation				
Balakovo Unit 4	Preliminary study	Qualitative study	Qualitative study	–
Novovoronezh Unit 6	Preliminary study	Preliminary study	–	–
Novovoronezh Unit 5	Preliminary study	Planned	–	–
Kalinin Unit 1	To be performed in 1997	To be performed in 1998	–	–

TABLE III. PSA STATUS OF WWER-440 REACTORS

Country/Plant	Level 1 PSA	Level 2 PSA	Level 3 PSA	Severe AM Guidance
Russian Federation				
Novovoronezh Unit 3&4	Preliminary study	–	–	–
Kola Unit 1	Performed	–	–	–
Kola Unit 3&4	To be performed in 1998	–	–	–
Slovakia				
J.Bohunice V-1 (230)	Performed ongoing revision close to finish	Started in 1996	–	–
J.Bohunice V-2 (213)	Performed	Planned	–	–
Mochovce (213)	Will start in 1997	–	–	–
Czech Republic				
Dukovany (213)	Performed	To be performed in 1997	–	Introductory study

Czech Republic

Probabilistic safety assessment and severe accident evaluation are not yet included in the Czech regulation. New rules, currently under preparation and soon to be implemented, will include PSA and severe accidents. Level 1 PSA has been carried out for both NPPs, Dukovany, WWER type 440/213, and Temelin, WWER type 1000. The Level 2 PSA for the Temelin NPP was completed in 1996 and the one for Dukovany will be in 1997. Severe accident studies were performed during the last five years. Initially the STCP code was utilized, now MELCOR and CONTAIN are used. The preparation of accident management procedures has been initialized.

The approach of the regulatory organization is based on the United States Nuclear Regulatory Commission (USNRC) generic letter for IPEs. For selected severe accidents there is the requirement to prepare accident procedures or measures. The selection is done based on PSA results, engineering studies and judgement and operational experience. After the elaboration of the severe accident measures and procedures, the specific sequences are re-evaluated.

The full scope Level 2 PSA study performed for the Temelin NPP resulted in the identification of the severe accident sequences which are dominant regarding consequences for the environment and population. The list of relevant severe accident scenarios based on the Level 2 PSA study was presented for discussion and review. On the basis of this list, the proposal for further activities in this field was put forward:

- To carry out detailed analyses of the dominant scenarios with the MELCOR code;
- To utilize the results of these detailed MELCOR calculations for evaluation of the proposed severe accident management measures;
- To perform off-site emergency planning;
- To further substantiate the assumptions and data for the Level 2 PSA study.

The Level 2 PSA study for the Dukovany NPP is under development, currently the construction of CET and their quantification has been completed. The sequences with the highest relevance were analysed with the MELCOR code. The results of these analyses show possible ways for improvements to the containment and containment systems.

Slovakia

A Level 2 PSA study has been started for the WWER-440/230 this year. The study will be carried out in co-operation with Nuclear Electric (UK), AEA Technology (UK) and two Slovak organizations (VUJE and RELKO). The progress of this study is delayed at present because it was necessary to revise the Level 1 PSA study. Investigation of the boundary conditions for the confinement are in progress at VUJE and the PDSs have been defined. The Level 2 PSA is considered to be very important for this type of plant because one of the most often commented weaknesses of the WWER-440/230 is that the units do not have a full size containment.

The Level 2 PSA for the WWER-440/213 plant is planned in the near future for the Buhunice V-2 NPP. The Mochovce NPP is currently under construction. Start of operation is planned for the end of 1998. The Level 1 PSA study should begin in 1997. Once Level 1 is completed, work on Level 2 PSA will commence.

Russian Federation

A qualitative Level 2 PSA study has been performed on the Balakovo Unit 4 NPP with the technical assistance of NNC (UK) and Belgatom (Belgium). Quantification of the consequences of containment event sequences is planned using deterministic analyses performed with the French ESCADRE code.

The following stages have been completed:

- Level 1, 2 interface;
- Development of the containment event tree and allocation of release categories to end states of CET;
- Containment probabilistic structural analysis using ABAQUS code and taking into account major uncertainties;
- Containment penetration and leakage analysis using an EPRI procedure;
- Determination of provisional source terms for each release category and their comparison with those given in NUREG-1150.

The following steps should be performed to finalize the study:

- To complete the severe accident analysis;
- To perform source term analysis;
- To quantify all CET sequences for all PDSs based on results of severe accident and containment fragility analyses.

The analyses are intended to show the need for improving containment performance.

5. IMPORTANT ISSUES

Important problem areas and issues regarding Level 2 PSA were discussed at the meeting. The description of these issues follows the order given in the IAEA's Level 2 procedures [1]. Issues which are specific to WWER type reactors and to channel type reactors are addressed in separate paragraphs.

5.1. GENERAL ISSUES

5.1.1. Scope and objectives of a Level 2 PSA

It was pointed out that the scope of the Level 2 PSA should be as complete as possible, including for example a complete set of initiating events, both internal and external, and all relevant plant operating states. A more comprehensive and complete assessment of human interactions should be considered in the Level 2 PSA.

Primary importance should be attached to prevention of severe accidents. Reducing the probability of containment bypass scenarios and improvements to the ability of the containment to withstand accident loads should receive particular attention. In addition, accident management activities should focus on the reduction of source terms, because they determine consequences and plant risk.

5.1.2. Organization and project management

It was pointed out that the support from the plant operator for the Level 2 PSA tasks is crucial. The requirements upon project organization and management for a Level 2 PSA differ significantly from those of a Level 1 PSA. Therefore it is important that the benefits of the Level 2 PSA be perceived by the plant operator. Furthermore, confidence in Level 2 PSA results must be established.

Technical and management interfaces for Level 2 PSA activities need to be defined in a particularly careful way. This includes the interfaces with the Level 1 PSA group and tasks, as well as with those of Level 3 if a Level 3 PSA is carried out.

5.1.3. Quality of the Level 2 PSA

A good Level 2 PSA quality is crucial for establishing confidence in the Level 2 PSA results. This requires first that a formal quality assurance programme be established and made effective for the Level 2 tasks. Such formal quality assurance programme includes items such as the definition of Level 2 PSA organization and management and procedures for the technical tasks to be carried out. Maturity and adequacy of approaches, codes, methods and data should be demonstrated. Additionally, different levels of review are important to obtain a quality Level 2 PSA.

5.1.4. Compilation of plant features, familiarization with the plant

Plant information used for Level 2 analyses should be as specific as possible. Therefore, the particularities of a specific containment should be known and taken into account when performing Level 2 analyses. Past analyses have shown that sometimes minor design and operational features may significantly impact on results. Plant walkdowns are particularly important for the Level 2 work. If the analysis is made for a plant in the design stage, assumptions may be necessary to perform the analysis, and should later be verified and revised, if necessary, by a detailed walkdown in the plant.

Analysts should have detailed knowledge of the plant. This is not a trivial requirement because analysts are often not directly affiliated with the plant. For example, the unknown existence of a single drain at the bottom of the reactor cavity can invalidate the results of extended analyses dealing with core concrete interactions.

5.1.5. Analysis of containment performance

Containment performance analysis determines the capacity of the containment and of its elements to withstand the various loads and conditions during accident sequences. This analysis should be comprehensive and realistic by including all important elements and factors, such as:

- Specific design and features of the containment to be analysed;
- Specific material properties;
- Influences of surveillance, tests, inspections, maintenance, repairs and effects of ageing;
- Initial and boundary conditions;
- Status of containment elements depending on plant's operating conditions and initiating and consequential events in accident sequences;
- Failure modes and the extent of failures;
- Dependencies;
- Loads and phenomena impacting on the containment during accident sequences;
- Performance of containment systems and equipment during accident sequences;
- Categorization and characterization of leaks and ruptures and other assumptions;
- Failure criteria;

- Conservatisms used when information is limited or missing;

- Availability and transferability of information from experiments.

Particular care should be given to containment bypass scenarios and to the analysis of containment isolation performance and reliability.

5.1.6. Accident progression analysis

It was pointed out during the meeting that the use of accident progression computer codes requires extended experience in order to avoid non-physical results and 'user effects'. The user needs to be aware of the code limitations in models and default values and specifications. The computer codes should be quality assured and validated as far as possible, in particular for their application to specific plant configurations and features.

Auxiliary and other plant buildings should be included in a realistic manner in the analysis if they have a potential to reduce the source terms. This corresponds to a comprehensive view of the concept of containment.

There was a general consensus of participants that accident progression codes and severe accident phenomena models need further development and research. Special attention should be given to code improvements for the long term phase of severe accidents and to their ability to model accident management measures. Specific severe accident phenomena which need further consideration are the following:

- Steam explosion, in- and ex-vessel, small scale steam explosions;

- Debris dispersion by small scale steam explosions, at vessel failure and debris-water interaction;

- Hydrogen deflagration and detonation;

- High pressure melt ejection;

- Core-concrete interaction;

- Direct containment heating.

There should be more activities for code validation and benchmarking of accident progression codes. More effort should be devoted to validate these codes as far as possible in order to increase the confidence in Level 2 results.

5.1.7. Development and quantification of the Level 2 PSA model

There was a broad consensus among the meeting participants that human behaviour and conditions for human interactions which have to be assessed in a Level 2 PSA need more attention and research. Human actions for accident management need to be supported by appropriate emergency procedures (including training, if implemented practically). Otherwise there can be very little confidence in the success of such actions. Assessment of human interactions should involve a careful investigation of conditions and influence factors.

Environmental qualifications for I&C, other important equipment and equipment used for accident management should be carefully evaluated. If expert judgment is used for bridging missing or limited knowledge or information it should be well structured, balanced and documented to establish confidence and traceability. Quantification of the models should be accompanied by the assessment of uncertainties and sensitivities for key assumptions and parameters.

The limitations of the current way of modelling the logic of event sequences for Level 2 PSA were discussed during the meeting. Today such modelling is usually done with logic event trees of various degrees of complexity. A majority of the participants declared that the present methodology is adequate for the purpose. However, for modelling long term sequences and sequences with complicated timing requirements, some participants pointed out that Markovian models might be more appropriate. Therefore, research in this area is recommended.

5.1.8. Uncertainty and sensitivity analysis

The performance of uncertainty and sensitivity studies is recommended for all Level 2 PSA studies. The outcome of the Level 2 PSA analysis depends in many aspects on modelling assumptions. This kind of variability should be investigated with sensitivity analyses which would help to establish confidence in the results. There is a need for guidance in systematic uncertainty and sensitivity analyses. Also, the methodology for assessing variability in models needs further development.

5.1.9. Interpretation of results

It was pointed out during the meeting that the interpretation of results has to include the results of uncertainty and sensitivity analyses. Uncertainty and sensitivity analyses are crucial for establishing confidence in the conclusions derived from the Level 2 PSA.

5.1.10. Issues related to the application of information and results

The uncertainty which exists for some aspects of Level 2 PSA is still a concern with respect to using the results in decision making. For the consideration of plant improvements and accident management actions and features, cost–benefit analysis can be performed and adverse effects of accident management actions and features should be considered. For the analysis of specific accident management actions and features, the PSA model sometimes needs adaptation or extension and special dedicated supportive analysis. Guidance for the performance of this kind of assessment is needed. During the meeting the question was raised whether and what kind of accident management actions or features need to be considered if the risk of a plant is already low. No comprehensive view on this aspect could be developed during the meeting.

5.2. WWER SPECIFIC ISSUES

5.2.1. WWER-1000

For WWER-1000 plants only two Level 2 PSA studies have been carried out to date on the following two NPPs:

- Balakovo NPP, Unit 4 (Russian Federation),
- Temelin NPP (Czech Republic).

Only a limited number of accident progression analyses have been performed for WWER-1000 plants. The Level 2 PSA work for Unit 4 of the Balakovo NPP considers several scenarios resulting from only one initiating event - a large LOCA. A full Level 2 PSA has been performed on the Temelin NPP. Similarly to the Balakovskaya NPP, only a limited number of accident scenarios were analyzed by the Rez Nuclear Research Institute and utilized in the study. Other required phenomenological information was obtained mainly from the literature (NUREG reports, other PSAs, IDCOR reports, etc.). Therefore, it is believed that the Level 2 PSA studies for the WWER-1000 plants have a significant level of uncertainty that can only be removed after more results from WWER-1000 specific analyses become available. Additionally, the results of the Level 2 PSA for the Temelin NPP are influenced by significant conservatisms resulting from a number of pessimistic assumptions used in the Level 1 part.

As a consequence, the two available Level 2 PSA studies should only be considered as preliminary. Nevertheless, the studies helped to increase the understanding of the major strengths and weaknesses of the containment, and allowed to pinpoint areas for more detailed investigations. Specific features of interest are the following:

- The containment bottom plate is located three levels above ground, the three levels in between are part of the non-hermetical auxiliary building.

- The reactor cavity is surrounded by 50 ionization chamber tubes that are located in the cavity walls very close (ca. 15 cm) to the inner surface of the cavity. These tubes penetrate the whole thickness of the containment basemat to the non-hermetical room beneath the containment. The tubes represent a potential weakness regarding penetration of the basemat by melted core material. In order to make reliable predictions, this item will need detailed investigation and analysis.

Analyses of the WWER pressure capacity performed for the Level 2 study show that the containment is robust and resistant to overpressure. With the exception of the specific basemat feature discussed above, it is comparable to other PWR containments. The following points were noted regarding a refined estimation of the containment capacity:

- Derivation of containment failure pressure should take into account all possible beyond design failure modes, i.e. catastrophic failure, failure of pressure boundary components (liner buckling or tearing, penetrations, insert plates, sumps, gaskets of airlocks etc.) with respect to effects of creep, corrosion and ageing.

- Uncertainty and randomness sources of corresponding containment fragility curves should be clearly identified and may be based on generic information or take into account plant specific information.

- A deterministic check should be performed for components of the pressure boundary to confirm their ability to withstand the ultimate pressure of global failure.

- For cases when the yield or buckling limit is being approached, the development of a probabilistic fragility curve is required.

- The possibility of dependencies between events should be looked into, such as between pressure rise and spray header malfunction or polar crane derailment.

5.2.2. WWER-440 (both versions, 230 and 213)

The confinement of the *WWER-440/230* is limited to hermetic rooms surrounding the primary circuit and the steam generators. The free volume of the confinement is relatively small. An assessment of the confinement response would need investigation of particular features such as overpressure capacity and an assessment of the locations and sizes of potential leaks. The confinement of the WWER-440/213 is equipped with a pressure suppression facility, the bubbler condenser. The free volume of the confinement is comparable with other PWRs. As for the 440/230, an assessment of the confinement response would need investigation of the overpressure capacity and the assessment of locations and sizes of potential leaks.

The main issues regarding analysis of confinement response for this reactor type are the following:

- Modelling of the confinement response with existing computer codes needs to take into account the particular features of these plants. The appropriateness and the limitations of the models in these codes need to be carefully ascertained.

- International knowledge and experience in this field are very limited.

5.3. SPECIFIC ISSUES FOR CHANNEL TYPE REACTORS

Two categories of issues for channel type reactors were identified during the meeting. There are many issues which are common to the general issues and shared with other types of plants. The following issues are specific for channel type reactors:

- Level 1 sequence end states, and as a consequence also plant damage states are different, unique and considerably more complicated for channel type reactors as compared to vessel type LWRs due to the different core design. Therefore, a different type of analysis is required to derive the PDSs for Level 2 PSA. An increased number of PDSs arises and their description is more complex.

- The study of accident progression is a key issue for the use of Level 2 PSA to evaluate and improve containment performance for this type of reactor.

Accident progression in the core region and in the whole plant has significant particularities for these reactor types. For PHWRs the time evolution of most of the severe accident scenarios is very slow due to the characteristic design features of the reactor. The most important design features of PHWRs in that respect are the following:

- large cooling capability of the moderator located around the fuel channels;

- additional enclosure provided by the calandria and calandria vault with large amounts of water;

- suppression pool and double containment in some designs.

While these specific design features may have a positive role in delaying and decreasing the severity of severe accidents progression, they lead also to increased difficulties and complexity in modelling of phenomena and processes.

There is a number of severe accident phenomena, processes and items which need to be studied in more detail for this type of plants, such as:

- moderator boiling;
- cooling effect of calandria vault water;
- core concrete interaction;
- suppression pool below calandria (in some designs).

6. FINAL REMARKS

It was suggested that a comprehensive report and a database containing descriptions of measures and features, envisaged to cope with severe accidents for different reactor types, be developed. For groups entering the Level 2 PSA and the field of severe accidents it appears to be difficult to obtain a systematic overview in this area.

It would be very useful to develop technical guidance on how to assess in practice accident management measures and features with PSA and other supporting studies. It would be also of interest to have procedures on how to develop design prescriptions for equipment and systems envisaged to cope with severe accidents.

Systematic and meaningful uncertainty and sensitivity analyses along with their interpretation is considered to be a difficult area which would require guidance. Uncertainty and sensitivity studies are recognized to be essential in Level 2 PSA to demonstrate the robustness of the Level 2 models and to establish confidence in the results. Guidance would be urgently required, in particular regarding sensitivity studies to deal with uncertainties from models and modelling assumptions. A further area which would need more attention is human reliability assessment for human interactions in Level 2 PSA.

Information exchange and feedback from other PSA teams working in the Level 2 PSA area are essential.

Due to the complexity of accident progression codes, organization of benchmark exercises and code workshops would be highly desirable.

Meeting discussions showed that the topic 'how safe is safe enough' remains very important and that guidance in this area is required, in particular regarding plant improvements through accident management.

WWER type reactors

Level 2 work for WWER type reactors is still in its beginning stages. It would be highly desirable that specialized meetings and comparison workshops be organized on this topic. Confinement performance analysis for the WWER type 440 (both 230 and 213) appears to be particularly difficult. The work done in this area is very limited, especially in the areas of structural analysis and analysis of leak locations and sizes. This information is a necessary prerequisite for a reasonably detailed Level 2 study.

Channel type reactors

It has become increasingly evident that detailed deterministic analysis of DBAs and beyond design basis accidents reveal considerable insights into channel reactor response to various accident conditions. These analyses can also help in better characterizing outstanding phenomenological uncertainties, improved EOPs and accident management strategies, including potential risk beneficial accident mitigative backfits. The analyses will also contribute to an increased understanding of the performance of the containment or accident localization system.

The PHWR group represented at the meeting formulated the following three specific recommendations:

(1) Modelling of severe accident phenomena for PHWRs should be a key point for further activities and the exchange of information for this type of plant.

(2) The proper functioning of equipment should be considered in Level 2 PSA, in particular with regard to its capability to work during severe accidents. The qualification of equipment may be upgraded, if necessary, from the design basis accident range to the conditions given during beyond design basis accidents.

(3) Level 2 PSA benchmark exercises would be especially helpful for the development of the methodologies for this type of NPP.

REFERENCE

[1] INTERNATIONAL ATOMIC ENERGY AGENCY, Procedures for Conducting Probabilistic Safety Assessments of Nuclear Power Plants (Level 2), Safety Series No. 50-P-8, IAEA, Vienna (1995).

LIST OF ABBREVIATIONS

AC	alternating current
AECL	Atomic Energy of Canada Limited
ALS	accident localization system
AM	accident management
ASDV	atmospheric steam discharge valve
ATWS	anticipated transient without scram
BDBA	beyond design basis accident
BWR	boiling water reactor
CANDU	Canada deuterium–uranium (reactor)
CCFP	conditional containment failure probability
CDF	core damage frequency
CET	containment event tree
CSDVS	condenser steam discharge valves
DCH	direct containment heating
DET	decomposition event tree
ECCS	emergency core cooling system
ESF	engineered safety features
FP	fission product
HEPA	high efficiency particulate air (filter)
HPCS	high pressure core spray
HPME	high pressure melt ejection
I&C	instrumentation and control
IPE	individual plant examination
LBLOCA	loss of coolant accident following a large pipe break
LOCA	loss of coolant accident
LWR	light water reactor
MDL	maximum design limits
MSLB	main steam line break
MSLBA	main steam line break accident
MUW	make-up water
NPP	nuclear power plant
PC	primary containment
PCCDS	primary containment controlled discharge system
PCFPB	primary containment filtration and pumpback system
PDS	plant damage state
PHTS	primary heat transport system
PHWR	pressurized heavy water reactor
PSA	probabilistic safety assessment
PSC	probabilistic safety criteria
PWR	pressurized water reactor
RBMK	light water cooled, graphite moderated channel type reactor
RC	release category
RCIC	reactor core isolation cooling
RCP	reactor coolant pump
RCS	reactor coolant system
RHR	residual heat removal
RPT	reactor pump trip
RPV	reactor pressure vessel

RSG	recirculating steam generator
SC	secondary containment
SCRPS	secondary containment recirculation and purge system
SDM	safety design matrix
SDCS	shutdown cooling system
SGTR	steam generator tube rupture
SLB	steam line break
SOL	safe operation limits
STCP	source term code package
TSG	technical support guidelines
WWER	water moderated, water cooled energy reactor

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Annex

PAPERS PRESENTED AT THE TECHNICAL COMMITTEE MEETING

XA9846917

A PROPOSAL FOR ACCIDENT MANAGEMENT OPTIMIZATION BASED ON THE STUDY OF ACCIDENT SEQUENCE ANALYSIS FOR A BWR

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Abstract

The paper describes a proposal for accident management optimization based on the study of accident sequence and source term analyses for a BWR. In Japan, accident management measures are to be implemented in all LWRs by the year 2000 in accordance with the recommendation of the regulatory organization and based on the PSAs carried out by the utilities. Source terms were evaluated by the Japan Atomic Energy Research Institute (JAERI) with the THALES code for all BWR sequences in which loss of decay heat removal resulted in the largest release. Identification of the priority and importance of accident management measures was carried out for the sequences with larger risk contributions. Considerations for optimizing emergency operation guides are believed to be essential for risk reduction.

I. Introduction

Accident management (AM) measures are to be implemented in all light water power reactors in Japan by the year 2000 [1]. Since the examination of accident management was requested to the utilities by the regulatory authorities, the Nuclear Safety Commission (NSC) and the Ministry of International Trade and Industry (MITI), each utility planned their strategy of implementing accident management in their power reactors [2], [3]. These measures were proposed on the basis of their probabilistic safety assessments (PSAs). They were approved after reviews by the regulatory bodies to be appropriate as a whole finally in November 1995.

JAERI is performing PSA studies for general purposes which include the examination of accident management. Analytical codes for evaluating core damage sequences are being developed and applied to power reactors for studying the consequences of severe accident sequences. Source term analysis with the THALES/ART [4] and THALES-2 [5] codes developed for this purpose is available, as source term models for release, transportation, deposition and revaporization of radioactive materials are incorporated in them. Since these codes run relatively fast, overall calculation of many sequences is feasible. Based on comparison of various sequences, numerous insights and knowledge concerning categorization of similar sequences could be gained.

With this knowledge and based on the study on the AM measures proposed by the utilities, the author proposes some important aspects which are desired to be incorporated in the emergency plant operation procedure guide for optimizing the effect of each AM measure.

II. Preceding Work

Kajimoto et al. [6] performed severe accident sequence analysis of a BWR and grouped the various sequences according to the scenario of pressure vessel failure and containment failure due to overpressure. They clarified the effects of core melt relocation, coupling of thermal-hydraulics and fission product (FP) vaporization, and containment failure location on the release fraction of CsI to the environment for a variety of sequences shown in Table 1 as sensitivity studies by using the THALES/ART (THALES-2) code.

The major findings of the work can be summarized as follows:

(1) The analyzed sequences initiated by transients and LOCA are basically categorized in 5 groups according to the similarity of the scenario from core damage to containment failure.

(2) The released mass of CsI also exhibits similar behavior in each group. It is affected by the melt relocation modes in which the core melt is once trapped at the core support plate at the bottom of the core or not. If the melt experiences high temperature there due to lack of coolant, the major part of FPs is released into the reactor coolant system (RCS) and deposits in the RCS, more or less, depending on the time duration in the sequence.

(3) Melt having not previously experienced high temperature may later release major amounts of FPs during the process to containment failure. This could result in larger source term release into the environment. An example of FP release in transient sequences is as given in Fig. 1 which shows larger release for such sequence groups having shorter FP deposition time like the TW groups named here Groups A and B, and the TC Group E.

(4) The effect of containment failure location on source term release is also significant as shown in Fig. 2 in which drywell space and wetwell liquid space failures with no water scrubbing effect show relatively large source terms.

(5) When thermal-hydraulics and FP revaporization are coupled in the code model, the revaporization of FPs in the reactor coolant system (RCS) during the containment depressurization process increases the source term to the environment by a factor of 10 at maximum compared with the source term without revaporization.

Watanabe et al. [7] performed further sequence analysis for the study of containment failure modes. They regrouped all significant accident sequences into 8 groups including interfacing system LOCA and reactor pressure vessel (RPV) rupture, and separated the loss of containment heat removal group into long-term and short-term as summarized in Table 2.

They obtained the conditional containment failure probabilities (CCFP) for various failure modes, for four separate accident progression stages in each sequence group and clarified the dominant containment failure modes and stages for each sequence group as summarized in Table 2. The dominant failure modes were mostly over-pressurization and over-temperature in this study. However, the dominating stage was different group by group. For example, containment failure due to overpressurization dominantly occurs in the 'pre-stage for core-melt' in Group 3 with loss of long-term containment heat removal, whereas containment failure occurs in the 'long-term progression stage' in Group 7 with loss of short-term containment heat removal. Though they also proposed mitigative measures by the use of conventional systems as shown in the table, these measures now should be replaced by the AM measures planned lately.

III. Discussions

The author reviewed the AM measures proposed by the utilities in the light of the correspondence with the 8 groups proposed in the previous work by Watanabe et al. and gave some consideration for optimizing those measures in timing and conditions of activation based on the above analytical results and other knowledge obtained through various experiments and analyses of severe accidents.

For a representative BWR-5 with Mark-II containment [1], [2] Table 3 summarizes the relation among each fundamental safety function, accident sequences contributing to core damage frequency (CDF) and containment failure probability and AM measures which are currently adopted and to be implemented together with the AM measures already implemented. The corresponding sequence groups G1 through G8 in the previous study are also indicated in the column of 'accident sequence' of this table. It can be confirmed that all significant sequence groups correspond to either of the individual AM measures in the table except for the interfacing system LOCA group, G1 which has a negligibly small contribution to CDF. Over-pressurization and over-heating scenarios of the containment which were dominant in most of the sequence groups in Table 2, are prevented or mitigated by the adopted AM measures such as 'alternative reactivity control', 'alternative water injection to reactor and containment', 'drywell cooler' or 'hardened vent'. Thus, all sequence groups can be covered by any of the AM measures already implemented and those adopted this time. The level of defence in depth for the core, pressure vessel and containment will be significantly increased by implementing all the AM measures.

Table 1. Accident Sequences Analyzed by Kajimoto et al. [6]

Transient			LOCA		
Transient (other than IORV)		IORV	Large Break	Medium Break	Small Break
TW	TB	T_iW	AW	S_1W	S_2W
TQW	TBU	T_iQW	AUW	S_1UW	S_2QW
TQU_1W	TBP	T_iQU_1W	AUV_1W	S_1UV_1W	S_2QU_1W
TQUW	TBPU	T_iQUW	AB	S_1B	S_2QUW
$TQUV_1W$	TQUX	T_iQUV_1W	AUV	S_1UX	S_2QUV_1W
TPW	TQUV	T_iB	AC	S_1UV	S_2B
TPQW TC		T_iBU		S_1C	S_2BU
$TPQU_1W$		T_iQUX			S_2QUX
TPQUW		T_iQUV			S_2QUV
$TPQUV_1W$					S_2C
TPQUX					
TPQUV					

A : Large break LOCA
B : Loss of all AC power
C : Failure to reactor scram
P : Failure to reclose relief valve
Q : Loss of feedwater
S1 : Medium break LOCA
S2 : Small break LOCA
T : Transient
Ti : Inadvertent opening of relief valve

U : Loss of all high pressure system
U1 : Loss of high pressure core spray
U2 : Loss of RCIC
V : Loss of all low pressure system
V1: Loss of low pressure core spray
V2: Loss of low pressure injection
W : Loss of residual heat removal
X : Failure to depressurization

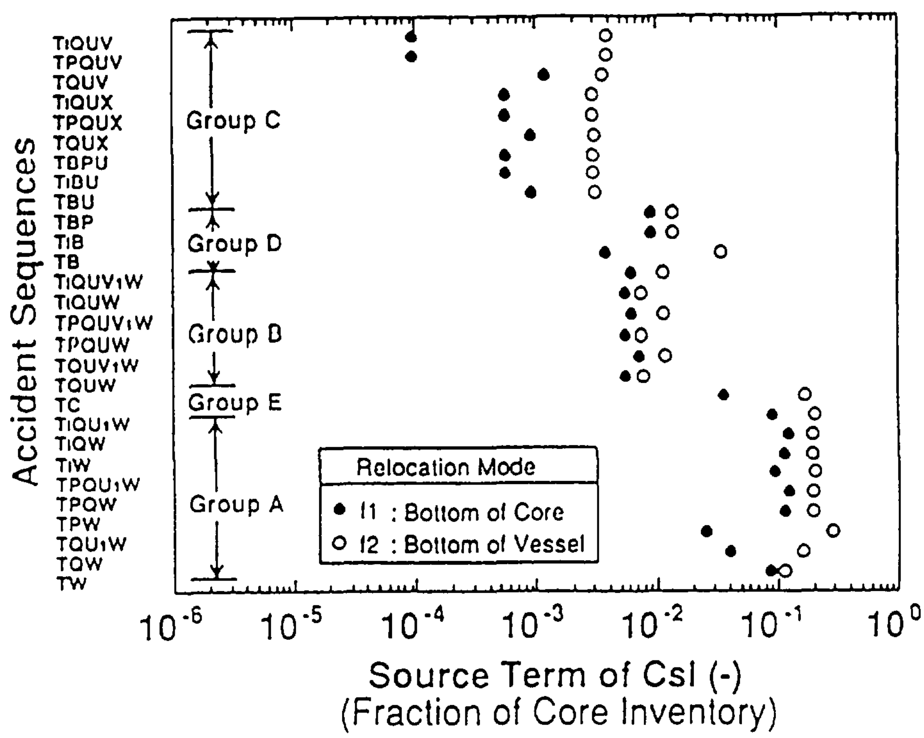

Fig. 1 Effect of Melt Relocation Modes on Source Term of CsI

For enhancing the effectiveness of these AM measures, some further consideration is necessary. Optimization of procedures is useful for establishing an effective emergency operation guide. Since risk is the product of CDF or resultant CCFP shown in Fig. 3(a) and 3(b) and the consequences of source terms, the importance of AM measures naturally focusses more on those which are effective for preventing and mitigating accident sequences with a larger risk contribution. It should be noted here that the contribution of each sequence to CDF and CCFP is different depending on the type of BWR as seen in Fig. 3 because of differences in system and safety function configuration. Nevertheless, there is a commonality for each type and therefore the discussion below will be mainly done based on the results for the BWR-5.

(1) Failure of Decay Heat Removal and Resulting Overpressure

One important recognition is that such sequence groups, having a dominant contribution to CCFP, as the residual heat removal system failure in a BWR-5 also shows a relatively large CsI release consequence as indicated in Fig. 1 and Fig. 2. Therefore the prevention and mitigation of those sequence groups, such as loss of containment heat removal (T--W) and anticipated transients without scram (ATWS or TC), have priority and need to be ensured for total risk reduction. The AM measures for containment heat removal are illustrated in Fig. 4(b). The 'hardened vent' is expected to ultimately ensure containment heat removal in the case the implemented 'manual operation of containment spray system' measure fails.

The determination of the optimum vent pressure is separated into two cases. In a sequence such as TW wherein containment overpressurization takes place earlier than core damage, the vent pressure can be set just above the maximum operating pressure of the containment to expedite the depressurization by releasing steam without FPs. In this case, confirmation of no core damage with radiation monitors and of the unavailability of containment spray and other cooling means is required before the operation of the vent system.

In the other overpressurization sequences which take place after core damage, a larger risk reduction can be achieved by setting the vent pressure as high as just below the endurable limit pressure of the containment and by extending the time of venting as long as possible allowing for FP decay. In this case, the 'drywell cooler' will contribute to suppressing pressurization and extending the time of venting. The wind direction may also be taken into consideration for risk reduction by a potential early venting in the process.

It is already known that water at saturated temperature reduces the scrubbing effect for FPs such as CsI [8]. Since, in the overpressure sequences, the water temperature is in saturation, minimum decontamination factors should be expected. Therefore, for deciding on the venting time, the temperature of the pressure suppression water must also be watched with instrumentation made reliable even in such an accident environment.

Recovery of heat removal is required in a long range in all cases. For determining the containment durability, demonstration tests are planned to be conducted by NUPEC [9]. This result will be useful for optimizing the venting conditions.

(2) ATWS

Alternative reactivity control measures such as reactor pump trip (RPT) and auxiliary control rod insertion (ARI) are adopted for the case of ATWS leading to core damage and containment overpressure. Although the contribution of ATWS to CDF is relatively small, source term release consequences of Group E (TC group) are large. The above measures will further ensure safe reactor shutdown and largely reduce risk. Since both of them are envisaged to be automatically activated in the case of the failure of the emergency scram signal, an operator is only needed to confirm the successful activation and for shutdown cooling afterwards.

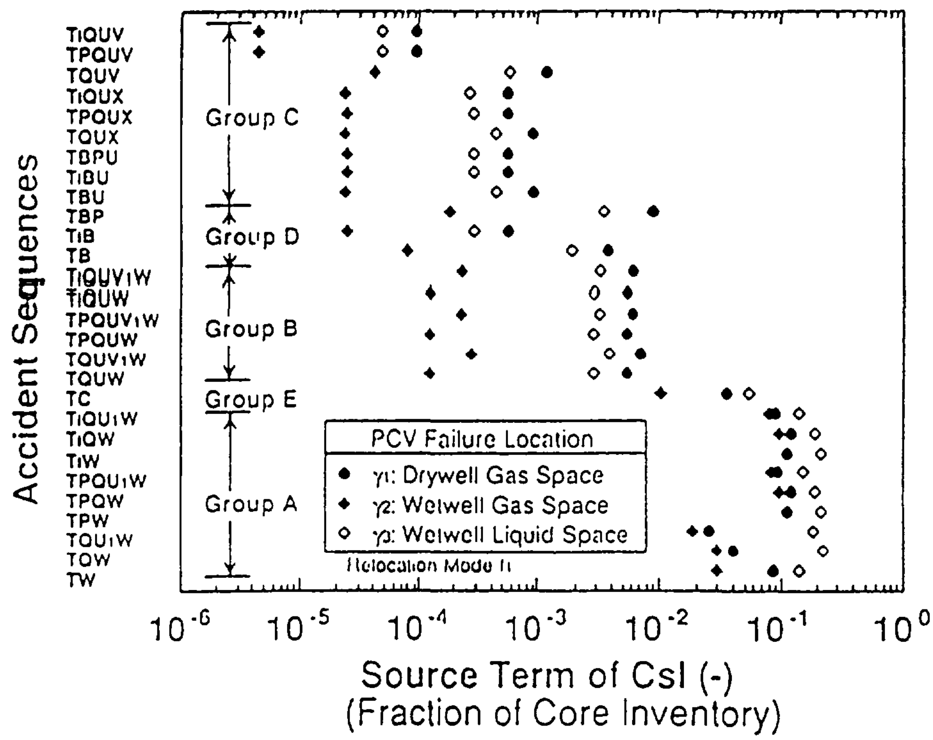

Fig. 2 Effect of Containment Failure Location on Source Term of CsI

(3) Loss-of-Electric Power Supply and Resulting Direct Containment Heating

In addition to the supply of high voltage electric power from the adjacent plant with a cross-tie already implemented, the 'supply of low voltage power' from the adjacent plant with another cross-tie and 'supply of power from D/G for HPCS' are adopted, as the BWR-5 is provided with a special D/G for HPCS. The operator who connects the cross-tie needs to confirm the activation of the D/G of the adjacent plant or of the D/G for HPCS for supplying power. The 'supply of low voltage power' from the adjacent plant will be used to activate D/Gs in case of battery system failure in the accident plant. All these measures for terminating SBO sequences, by establishing the emergency guide and training of the operators, will contribute to eliminate the DCH sequence in Fig. 3(b) and the large source term release of Group D (TB sequences).

(4) Failure of Water Injection or Depressurization and Resulting Over-heating

The contribution of those sequence groups initiated by failure of water injection and depressurization of RPV to CDF is relatively small. The consequential source term release mass is also the smallest of all the groups. However, the probability and consequence of a steam explosion still have a large uncertainty. If it occurs violently in the pressure vessel or in the containment, the source term mass release out of the containment drywell may significantly increase as can be seen in Fig. 2 for Group C.

It is already known that dropping of the melt in water at relatively high pressure (> 1MPa) or with a temperature near saturation does not cause significant steam explosion [10]. Therefore, water mass distribution and the pressure and temperature of the water are crucial for the prediction of steam explosion occurrence. Melt relocation and water conditions must be carefully monitored with required instruments to avoid a steam explosion. When water injection is required for melt cooling both in the pressure vessel and containment, spraying water on the melt is more desirable than filling plenty of water in a place beneath the melt where it finally slumps (see Fig.4(a)). Effective melt cooling conditions also need to be established through research and reflected in the emergency guide to prevent the melt spreading on pedestal floor and directly attacking the containment shell as well as the concrete floor.

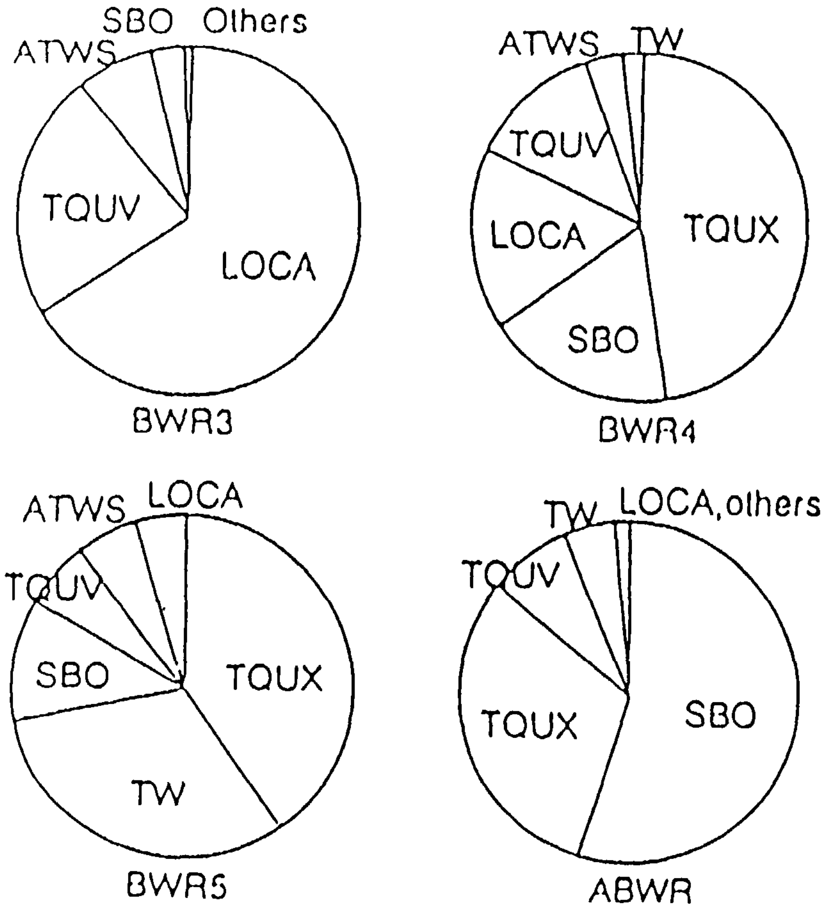

Figure 3(a) level-1 PSA results [2]

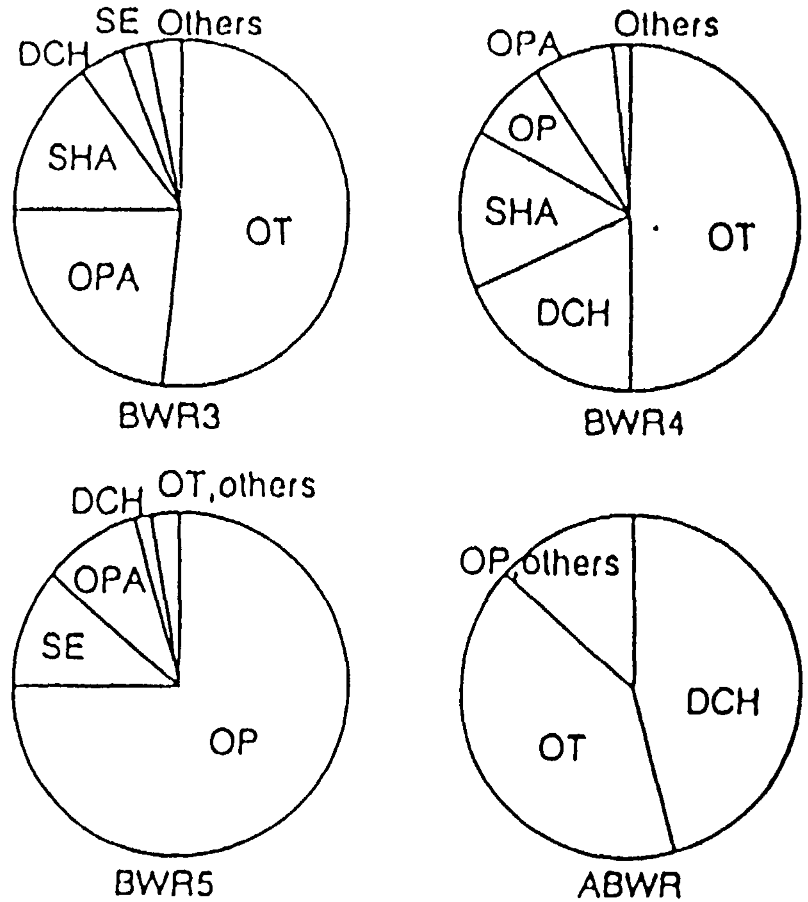

Figure 3(b) level-1.5 PSA results [2]

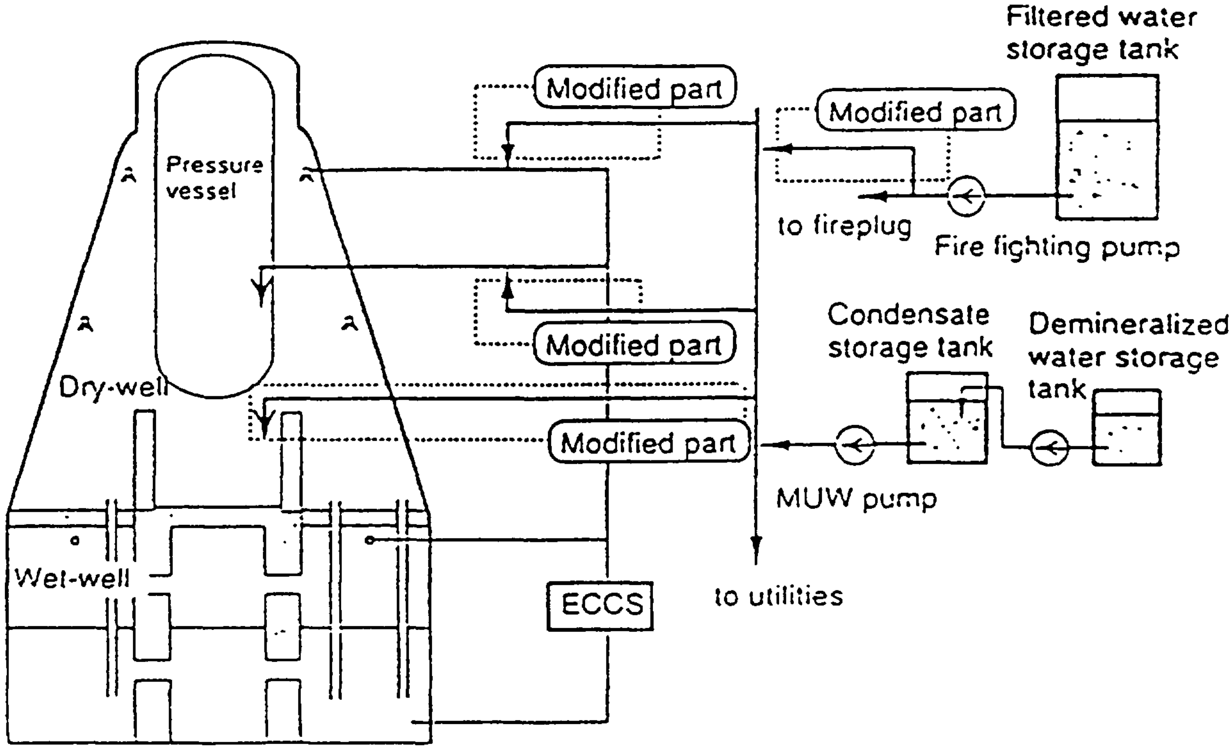

Fig. 4(a) Alternative water injection [1]

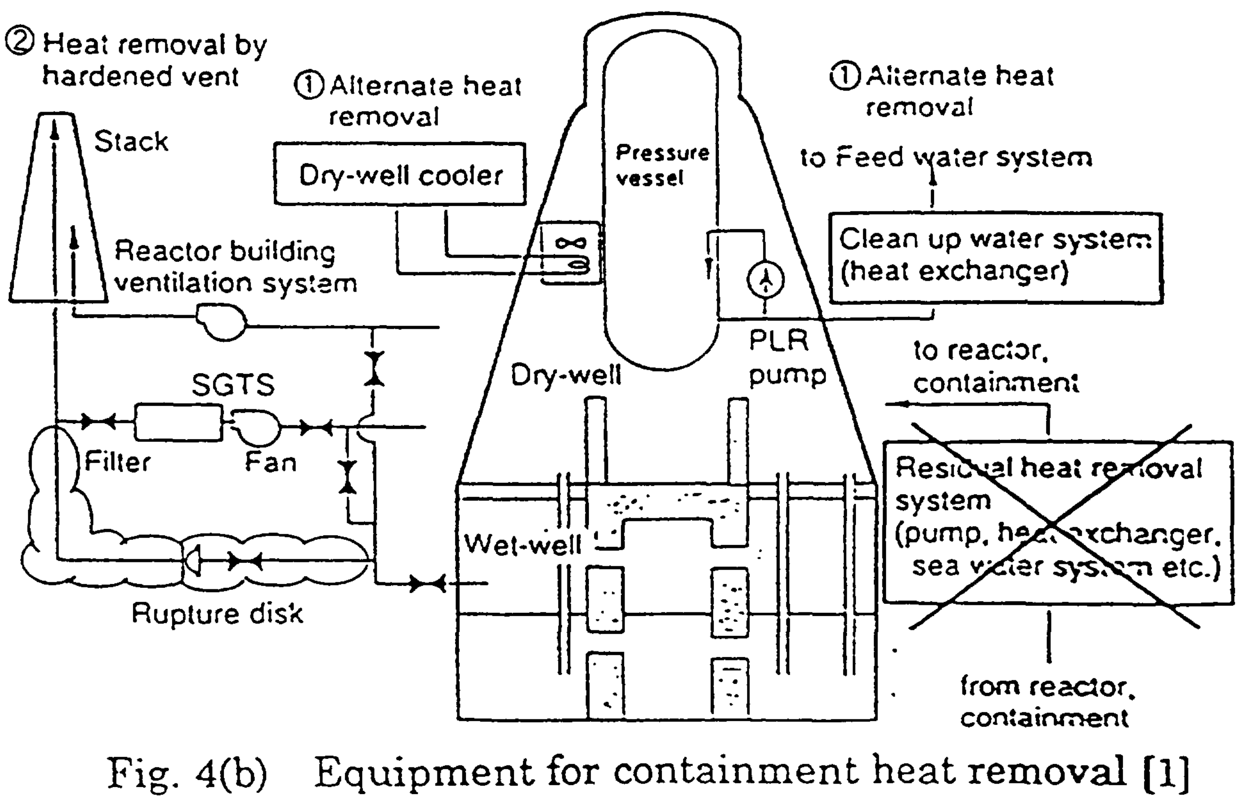

Fig. 4(b) Equipment for containment heat removal [1]

Table 2. Summary of Sequence Categorization by Watanabe et al. [7]

Sequence Group	Core Damage Sequences	Dominating Failure Mode	Effectiveness of Mitigative Measures Considered
Interfacing System LOCA (Group 1)	V-sequence	Containment Bypass (Pre-stage for Core-melt)	Not Applicable
ATWS with High Pressure Injection Available (Group 2)	TC,S2C,S1C,AC	Overpressurization (Pre-stage for Core-melt)	Not Applicable
Loss of Long-term Containment Heat Removal (Group 3)	TW,TQW,TQU1W,TPW, TPQW,TPQU1W,S2W, S2QW,S2QU1W,S1W,AW	Overpressurization (Pre-stage for Core-melt)	Containment Venting (Pre-stage for Core-melt)
Station Blackout (Group 4)	TB,TPB,TBU,TPBU	Overpressurization Over-temperature (Long-term Prog.Stage) Direct Containment Heating (Debris Exit Stage)	Off-site Power Recovery (Core-melt Prog.Stage) (Long-term Prog.Stage)
Loss of Coolant Injection with Reactor Not Depressurized (Group 5)	TQUX,TPQUX,S2QUX, S1UX,TCU,S2CU,S1CU	Overpressurization (Core-melt Prog.Stage)	Reactor Depressurization (Core-melt Prog.Stage) Debris Cooling by RHR (Long-term Prog.Stage)
Loss of Coolant Injection with Reactor Depressurized (Group 6)	TQUV,TPQUV,S2QUV, S1UV,AUV	Overpressurization Over-temperature (Long-term Prog.Stage)	Not Applicable (It would be possible to repair the failed RHR components in Long-term Prog.Stage)
Loss of Short-term Containment Heat Removal (Group 7)	TQUW,TQUV1W,TPQUW, TPQUV1W,S2QUW, S2QUV1W,S1UW,S1UV1W, AUW,AUV1W	Overpressurization Over-temperature (Long-term Prog.Stage)	Not Applicable (It would be possible to recover the containment cooling in Long-term Prog.Stage)
Reactor Pressure Vessel Rupture (Group 8)	R-sequence	Overpressurization (Pre-stage for Core-melt)	Debris Cooling by RHR (Long-term Prog.Stage)

Table 3. Accident Management Measures of the Representative BWR

Function	Accident Sequence :Corresponding Group	Up: AM Measures Adopted This Time Lo: AM Measures Implemented
Reactor Shutdown Function	-ATWS : G2 =Overpressure after ATWS : G2	>Alternative reactivity control
		>Manual operation of boric acid injection system
Water Injection into Reactor and Containment	-Failure of high pressure water injection and depressurization : G5 -Failure of high and low pressure injection : G6 =Over-heating of penetration (steam explosion) : G7	>Automatic depressurization by the signal of transient (low reactor water level) >Alternative water injection (water injection to reactor and containment using pumps with condenser makeup water system or fire fighting system)
		> Alternative water injection (>Water injection to reactor through feed water system or control rod drive hydraul. system >Water injection to reactor and containment by sea water system)
Heat Removal from Containment	-Failure of decay heat removal : G3, G7 =Overpressure by steam (decay heat) : G2- G8	>Drywell cooler >Hardened vent
		>Manual operation of containment spray system
Support of Safety Function	-Loss-of-electric-power supply : G4 =Direct containment heating : G4	>Supply of electric power (cross-tie of power supply in low voltage from adjacent plant and high voltage from D/G for HPCS)
		>Supply of electric power (cross-tie of power supply in high voltage from adjacent plant)

IV. Summary

Through the review of the source term analysis study, the sequence grouping study and recent experimental knowledge in severe accident phenomena, the relativity of the importance of AM measures adopted for BWRs was discussed. Some proposals to be considered in emergency operation procedures were made in the viewpoint of the reduction of risk as the product of CDF and source term release consequences.

This examination also addressed the area where further clarification in the phenomena for reducing uncertainty is required to monitor accident progression in a reliable way and for taking appropriate measures for mitigating consequences. Although the present analytical models and results include some uncertainty, general insights obtained through this review are considered to be valid.

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XA9846918

PARAMETRIC MODEL TO ESTIMATE CONTAINMENT LOADS FOLLOWING AN EX-VESSEL STEAM SPIKE

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Abstract

This paper describes the use of a relatively simple parametric model to estimate containment loads following an ex-vessel steam spike. The study was motivated because several PSAs have identified containment loads accompanying reactor vessel failure as a major contributor to early containment failure. The paper includes a detailed description of the simple but physically sound parametric model which was adopted to estimate containment loads following a steam spike into the reactor cavity.

1. INTRODUCTION

Several Probabilistic Risk Assessments (PRA) have identified containment loads accompanying reactor vessel failure as a mayor contributor of early containment failure during severe accidents. Relatively simple, parametric models have been developed to which allow the PRA analyst to evaluate the range of conditions under which the containment loads may be an important contributor to the containment failure. In this paper a set of calculations utilizing those parametric models to estimate containment loads for Laguna Verde Nuclear Power Plant are presented, as well as a discussion of its utilization in the PRA.

2. DESCRIPTION

The Laguna Verde Nuclear Power Plant (LVNPP) Unit 1 is a 1931 Mwt BWR-5 boiling water reactor, designed and supplied by the General Electric Company with a Mark II containment.

The containment design employs a steel lined reinforce concrete structure utilizing the over-and-under pressure suppression concept. The pressure suppression concept consist of a drywell over a pressure suppression chamber or wetwell. The drywell and wetwell are separate for a concrete floor (diaphragm floor) which is perforated by 68 downcomer pipes to drive the steam released during a Loss Of Coolant Accident into the suppression pool. To suppress the pressure in the containment during an accident, two trains of sprays are located in the containment. In the event of failure of sprays to suppress the pressure, the containment can be vented. To reduce the potential of a severe hydrogen combustion event the containment is inertized with nitrogen.

3. PSA Requirements

Several Probabilistic Risk Assessments (PRA's) have identified containment loads accompanying reactor vessel failure as a major contributor of containment failure during postulated severe accidents[1]. One potential contributor to those loads is a phenomenon referred to as "*steam spike* ". A Steam spike is a rapid (but non-explosive) generation of steam that results from the interaction of molten core debris with a pool of water.

In the reactor cavity of the Laguna Verde Nuclear Power Plant (LVNPP) are located the control rod drive mechanism along wiht two pools to storage the floor and equipment drains (3.96 m^3 of storage capacity each). The liquid collected in those pools are small leakages of the system components during normal operation. In the case of a containment sprays actuation, there would be a significant amount of water in the above mentioned pools. Therefore ,if the bottom of the reactor vessel fails, the molten core debris will fall down into the reactor cavity and a steam spike phenomena would be expected.

The PRA team have to analyse, the likelihood that the steam spike phenomena occurs for each sequence related with reactor vessel breach. Analytical methods for estimating containment loads have focused on the development of mechanistic models for integration in large computer codes such as CONTAIN[2]. However, the application of quite sophisticated mechanistic models requires an impractical investment on time and financial resources to analyse hundreds of accident sequences of the containment event tree. The methodology utilized in the NUREG-1150[1] lets the PRA analyst to handle parameters and probabilities for the containment event tree heads and this flexibility then allows the analysis of several accident progression pathways for a fixed set of initial condition included in the definition of the plant damage state (i,e. Is highly uncertain the amount of zircaloy oxided before vessel rupture, so that a set of possible answers is required) . In order to provide a practical tool, we propose to use a relatively simple but physically sound model to represent the mass and energy addition to the containment following core meltdown and breach of reactor vessel.

4. MODEL DESCRIPTION

Models for steam spike and direct containment heating developed by Mark T. Leonard[3] are particularly convenient for the present analysis .The models are simple, physically sound and were developed for Boiling Water Reactors.

The model considers three steps:

1. Molten core material emerges from the reactor vessel and falsl into a pool of water that covers the drywell floor, for LVNPP we have to consider the equipment drains collector sumps. The rate, composition and temperature at which melt would leave the reactor vessel, are highly uncertain and several possibilities were examined parametrically. A wide range in values for the following initial conditions were considered:

   - total mass of molten material interacting with water.
   - initial temperature of the melt.
   - fraction of zirconium mass that was oxidized in-vessel.

2. When the melt reaches the water pool, rapid steam generation facilitates mixing of fragmented melt particles with water. The extent of melt-coolant mixing is limited by the rate at which steam can escape the water pool. Unmixed melt mass is assumed to fall to the bottom of the water pool and not be quenched.

3. Molten debris transfers latent and sensible heat to the water pool. Oxidation of molten zirconium/steel metal by sorrounding water consumes steam but produces hydrogen in equal molar proportion. The reactions are also highly exothermic, adding to the melt sensible heat that can be transferred to the water.

The maximum extent of heat transfer to the water and chemical energy addition to the melt from oxidation reactions over the time frame of interest are rate limited. In the case of heat transfer to water, heat losses from fragmented debris (melt) particles are limited by film boiling at the particle surface. The Production of chemical energy is limited by oxidation kinetics for metalic debris constituents.

As the debris mass falls down into the pool, it breaks apart into smaller particles which may subdivide further as steam produced at their surfaces flowing upward toward the surface of the pool. Melt fragmentation continues until the rate of steam production precludes further debris-water contact (i.e., the remaing water is fluidized and transported away from unfragmented debris) or the falling debris mass reaches the bottom of the pool.

Following reference [3] the limit to melt-coolant mixing is defiened as a coolant fluidization (countercurrent flow) limit. The maximum mass of molten debris that can be mixed with the coolant is estimated as

$$\frac{6m_{d,\max}}{\rho_\ell D_{mix}} q''_{part} = \rho_v h_{fg} A_{ch} V_{sc} \quad , \tag{1}$$

where

$m_{d,\max}$ maximum (mixed) debris mass in coolant pool
ρ_ℓ, ρ_v liquid and vapor (water) density, respectively
D_{mix} characteristic diameter of mixed debris particles
q''_{part} heat flux at the particle surface
h_{fg} latent heat of vaporization for water
A_{ch} drywell floor area (pool cross-sectional area)
V_{sc} superficial critical vapor velocity (i.e., steam velocity that will fluidize entering liquid coolant)

The total extent of (metal) particle oxidation during the melt/coolant interaction is treated parametrically (i.e. specified ahead of time). The total mass of molten debris poured into the coolant pool is also specified as a parametric variable.

The energy of reaction corresponding to the specified extent of melt oxidation is added to the enthalpy of the debris before calculating the maximum mass of melt that can be mixed with the coolant. However, the mass of oxidized melt is assumed to be limited by the mass of melt that can be mixed with the coolant.The maximum mass of melt that can be mixed with coolant is a function of debris temperature. Therefore, an interative solution is required.

1. an estımatation of the fraction of the poured melt mass that mixes with the coolant of the pool (the remainder is assumed to sink to the bottom of the pool uncolled).

2. the film boiling heat flux at the surface of a melt particle of characteristic dimension

D_{mix}.

The total amount of energy transferred to the coolant pool is then simply the product of the heat flux at the particle surface (q''_{part}) and the total surface area represented by the mixed mass of melt

$$Q_{pool} = q''_{part} N_{part} \pi D_{mix}^2 \Delta t \quad , \tag{2}$$

where Δt is the time interval in which the steam generation produces an increase in drywell pressure (1 s).

The total number of debris particles of characteristic size D_{mix} is given by

$$N_{part} = \frac{6}{\bar{\rho}_d \pi D_{mix}^3} \min[m_{pour}, m_{d_{max}}] \tag{3}$$

where

m_{pour} mass of melt poured into coolant pool

$m_{d,max}$ maximum mass of debris that can mix with the coolant pool

Assuming the pool is initially at the saturation temperature, the mass of water vaporized can be estimated. However, as discussed earlier, some of the steam is consumed in the oxidation of a (predefined) fraction of the metallic melt. Two moles of hydrogen are produced for every mole of steam consumed in oxidizing Zircaloy, and (using iron to represent the reactions of stainless steel) one mole of hydrogen is produced per mole of steam consumed in oxidizing steel. If the fraction of metal constituents in the molten pour oxidized ex-vessel is defined as f_{ox} and the fraction of poured mass that is Zircaloy is defined as f_{zr}, then the total number of moles of vapor generated is

$$N_{spike} = \frac{Q_{pool}}{h_{fg} Mw_{(H_2O)}} (1 - f_{zr} f_{ox}) \quad , \tag{4}$$

where

$Mw_{(H_2Ol)}$ is the molecular weight of water.

The initial number of moles in the drywell atmosphere ($N_{RC,i}$) can be estimate from the ideal gas law:

$$N_{RC,i} = \frac{P_i V_{RC}}{R_0 T_i} \quad , \tag{5}$$

where

p_I pressure of the drywell at vessel breach
V_{RC}. total free volume of the drywell
R_0 universal gas constant
T_I temperature of the drywell atmosphere at vessel breach

Assuming an adiabatic (isothermal) pressure transient, the incremental pressure due to steam spike and ratio of peak to initial pressure are

$$p_f - p_I = \frac{N_{spike} R_0 T_I}{V_{RC}} \tag{6}$$

and

$$\frac{P_f}{P_I} = \frac{N_{spike} + N_{RC,I}}{N_{RC,I}} , \tag{7}$$

respectively,

5. RESULTS

In order to show the estimates obtained with the above parametric model, we present two sets of calculations. In the first set, we assume

- A particle diameter of 10 mm
- 70% of the zirconium was oxidized in-vessel
- 1700 K of initial melt temperature.

The second set assumes a particle diameter of 100 mm with the same overall assumption as in the first set.

Figure 1 shows that the smaller particle size precludes faster the premixing phase because as the particle size becomes smaller the surface area for heat transfer become higher, therefore the steam generation increase impeding the interaction between molten core debris and coolant.

Figure 2 and 3 show the tendency to reach an increase pressure limit independent of the molten core debris poured. The limit is bounded by the maximum mixed mass and the transient time. The steam generated beyond the transient time (1s) is assumed to be fully condensed. Figure 2 shows, in the 10% and 30% cases that the pressure increases, this is due to an increase in melt temperature, and therefore the pressure reaches its maximum, as the melt temperature fits the temperature at which the 10% and 30% of the mass is mixed, as showned in the figure 1. Figure 3 shows the same behaviour as the figure 2 but the profile remains unchanged due to that the fact that the maximum mixed mass is not reached .

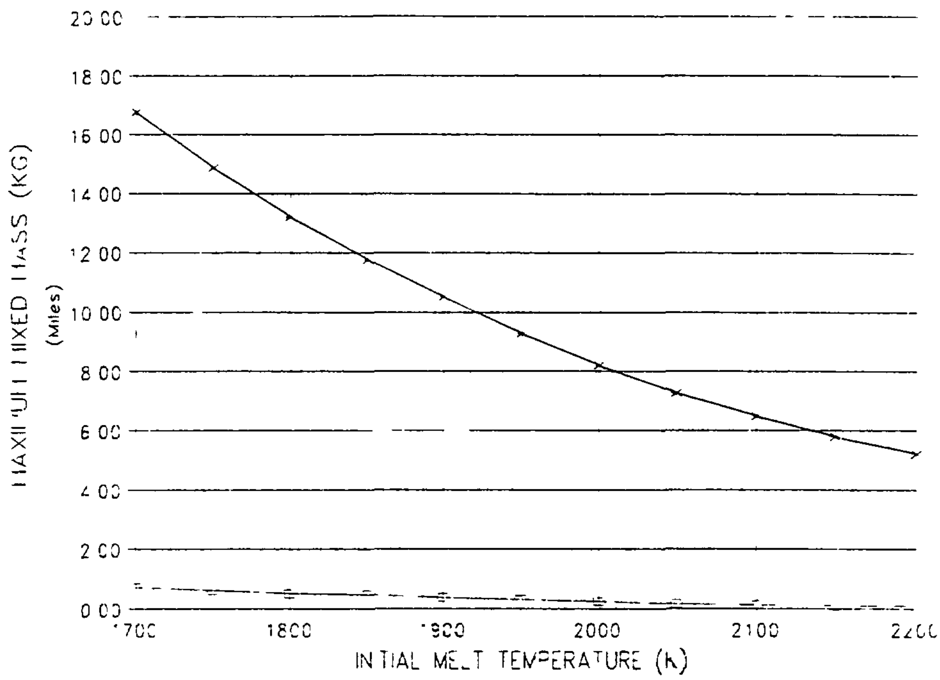

FIGURE 1. Maximum mixed mass for a 10 and 100mm.

* 100mm
o 10mm

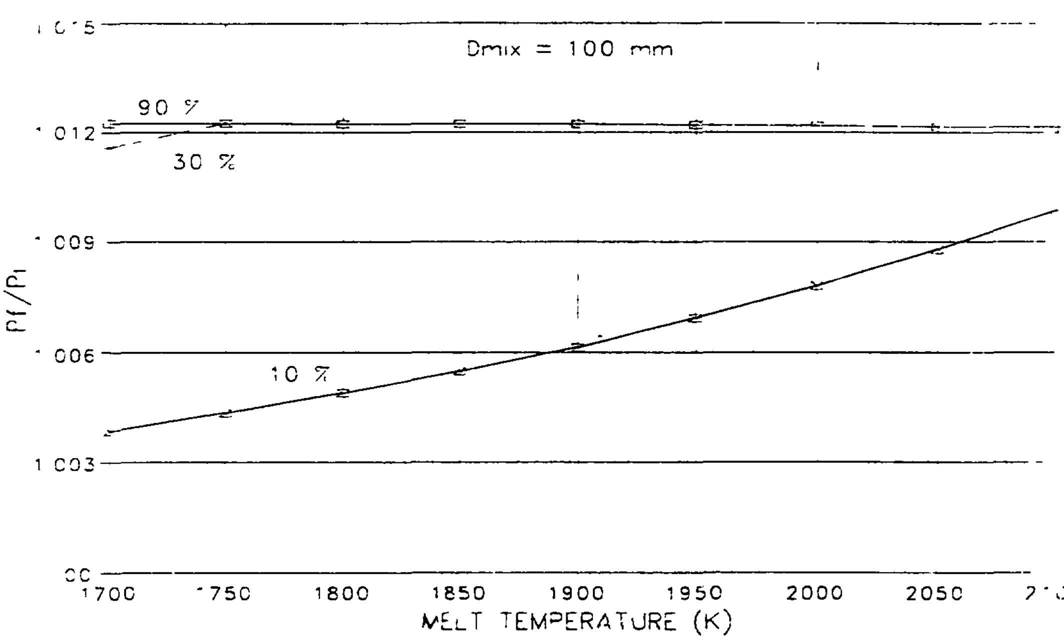

FIGURE 2. Ratio of peak to initial pressure for the 10, 30 and 90% of the total metalic mass .

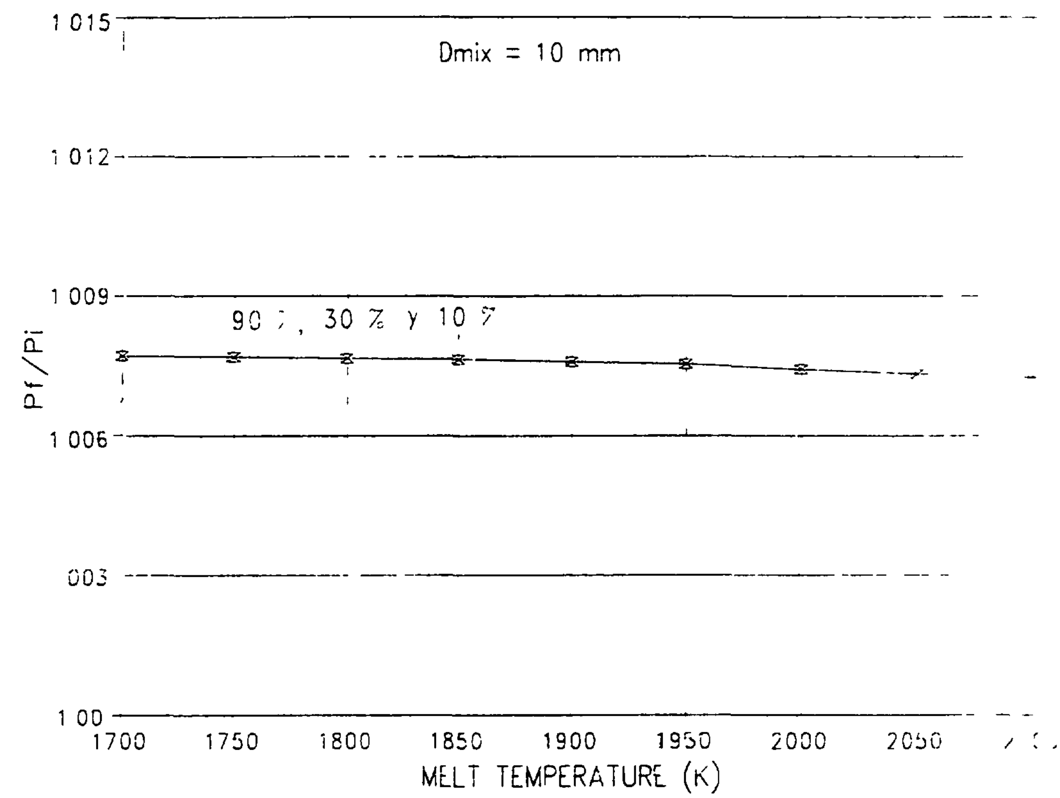

FIGURE 3. Ratio of peak to initial pressure for the 10, 30 and 90% of the total metalic mass.

6. CONCLUSIONS

The relatively simple but physically sound model adopted here allows the PSA analyst to perform speedy parametric (sensivity) studies of the contribution of steam spike containment loads.

The model described in this paper is not intentded to replace mechanistic models for steam spike, rather, as a tool to provide rough estimates.

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STATE OF LEVEL 2 ANALYSES AND SEVERE ACCIDENT MANAGEMENT IN SPANISH NUCLEAR POWER PLANTS

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Abstract

The state of the PSA/IPE studies in the Spanish NPPs is presented in this report, as well as the plans to implement the severe accident management strategy both in the Spanish BWRs and PWRS. First, the Spanish LWRs are introduced, and the scope of the IPE analyses required by the Spanish Regulatory Commission (CSN) is given. The general overview is completed with the current degree of development for the IPE studies in each plant.

In the second part the methods and tools are shown which are used by the Spanish plants to develop their Level 2 analysis. The different strategies for severe accident management adopted by the BWRs and PWRs in Spain are also outlined. The sources and implementation of the Severe Accident Guidelines (SAG) are described.

More detail is given in the following chapters to the containment analysis of Trillo (PWR, KWU design) and Cofrentes (BWR/6, GE design) NPPs, whose development is being carried out by IBERDROLA. The analysis which has been performed up to date for Trillo is limited to the Plant Damage State (PDS) definition. However, the main phenomena challenging its containment performance have been identified, and they are summarized here. The Level 2 analysis for Cofrentes is comparatively more developed. The main phenomena and the key equipment affecting its containment behaviour are presented. Finally the conclusions of this report are elaborated.

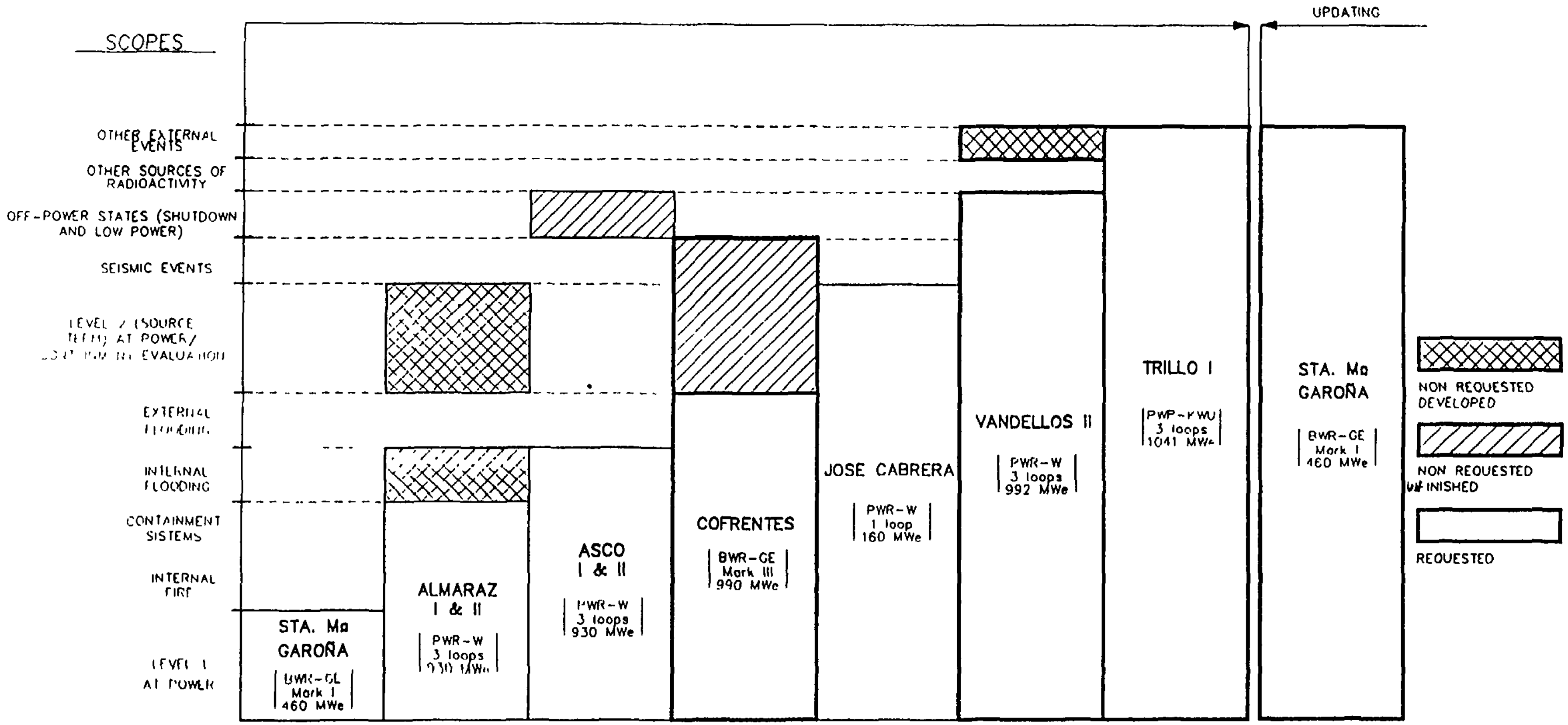
1.- STATE OF PSA/IPE DEVELOPMENT IN SPAIN

There are several types of LWR in Spain: from BWRs, General Electric design with Mark-I and Mark-III containments, to PWRs of Westinghouse (6 units, one of them with 1 loop) and KWU design (see Fig.2.1). The Spanish Regulatory Commission (CSN) started to require the Utilities an increasingly extended scope for their PSAs, beginning in 1984 by demanding a Level-1 scope to the first one (S^{ta} M^{a} de Garoña), and requiring a full IPE (Level-1, Level-2, IPEEE. and off-power states) to C.N. Trillo-1, the last LWR put into operation in Spain (see Fig.2.1). CSN's plans are to request a full scope of the IPEs in each plant, extending their previous analyses.

The current state of the Level-2 analyses in the different commercial LWRs in Spain is the following:

BWRs

- CN Cofrentes (GE, Mark-III) Level 2 under development
- CN Garoña (GE, Mark-I) Level 2 being started, to be finished in 1997



84	JUL 86	JUN 87	FEB 88	JUL 89	OCT 90	JUL 91	ENE 95	CSN REQUIREMENT
	APR 87	JAN 89	FEB 89	JUN 90	JUL 91	JUL 92	--	"PROJECT PLAN" APPROVAL
MAR 84	MAR 87	DIC 88	FEB 89	NOV 90	SET 91	SET 93	--	PSA STARTING
SET 85	NOV 89	SET 92	JUL 90	JUL 93	OCT 94	--	--	PSA rev. 0 ENDING
FEB 87	APR 90	--	MAR 95 (prev.)	--	--	--	--	PSA rev 1 (including CSN's comments)

FIG. 2.1 PSAs IN SPAIN NPPs
SCOPES

PWRs

- CN Almaraz (2 units; Westinghouse-3 loops) Level 2 prepared, although not requested yet
- CN Vandellós-2 (1 unit; Westinghouse-3 loops) Level 2 under CSN revision
- AN Ascó (2 units; Westinghouse-3 loops) Level 2 not started
- CN José Cabrera (1 unit; Westinghouse-1 loop) Level 2 under CSN revision
- CN Trillo (1 unit; KWU-3 loops) Level 2 under preparation

2.- GENERAL TRENDS IN SPANISH NPP

2.1.- METHODS AND TOOLS

The methodology requested by the CSN to be used in the Individual Plant Examination of the Spanish LWRs is the well known collected in the US-NRC Generic Letter 88-20. All the Spanish Utilities performing Level 2 analyses have followed the method consisting in a PSA plus a Containment Performance Analysis, as advised in Appendix 1 of that Generic Letter.

MAAP3 is the tool being used by all the plants to simulate the severe accident progression and to perform the sensitivity analyses on uncertain phenomena. However, some of the plants have developed models and expertise with the MAAP4 and CONTAIN codes, although these codes are planned to be used only in particular cases if need exists. MAAP4 models are being prepared by some plants as a support tool to the implementation of the Severe Accident Management guides.

A general policy of the plants is to use the results of their Containment Analysis to justify the non installation of new Containment equipment. Nevertheless, some of them had got some equipment installed before starting their analysis, mainly because of anticipation on previous American BWR experience (see §4 below).

2.2.- BWRs SEVERE ACCIDENT MANAGEMENT

The Owners Group of Spanish BWRs has decided to implement into their organizations and procedures a common strategy for the management of severe accidents. That strategy is based on the one of its American counterpart. It is collected into the Accident Management Guidance (AMG). The AMG includes the Emergency Procedure Guidelines (EPG) and the Severe Accident Guidelines (SAG). The SAGs are conceived as a continuation of the EPGs, and changes are introduced to smooth the transition from one guide into another. The Technical Support Guidelines (TSG) are also merged into the AMG to provide engineering assistance to the emergency organization. Finally, the AMG includes the criteria to assign the responsibility of given actions out of the Control Room.

The Containment Analysis of the Spanish BWRs has been oriented from the beginning to training. The results of all the simulations performed for the IPE are included into a library. Suitable tools will be available to retrieve from this library a selected scenario, with the explanations of the situation and the evolution of its plant conditions. The Spanish BWROG is considering also the convenience of simulation tools for severe accident training.

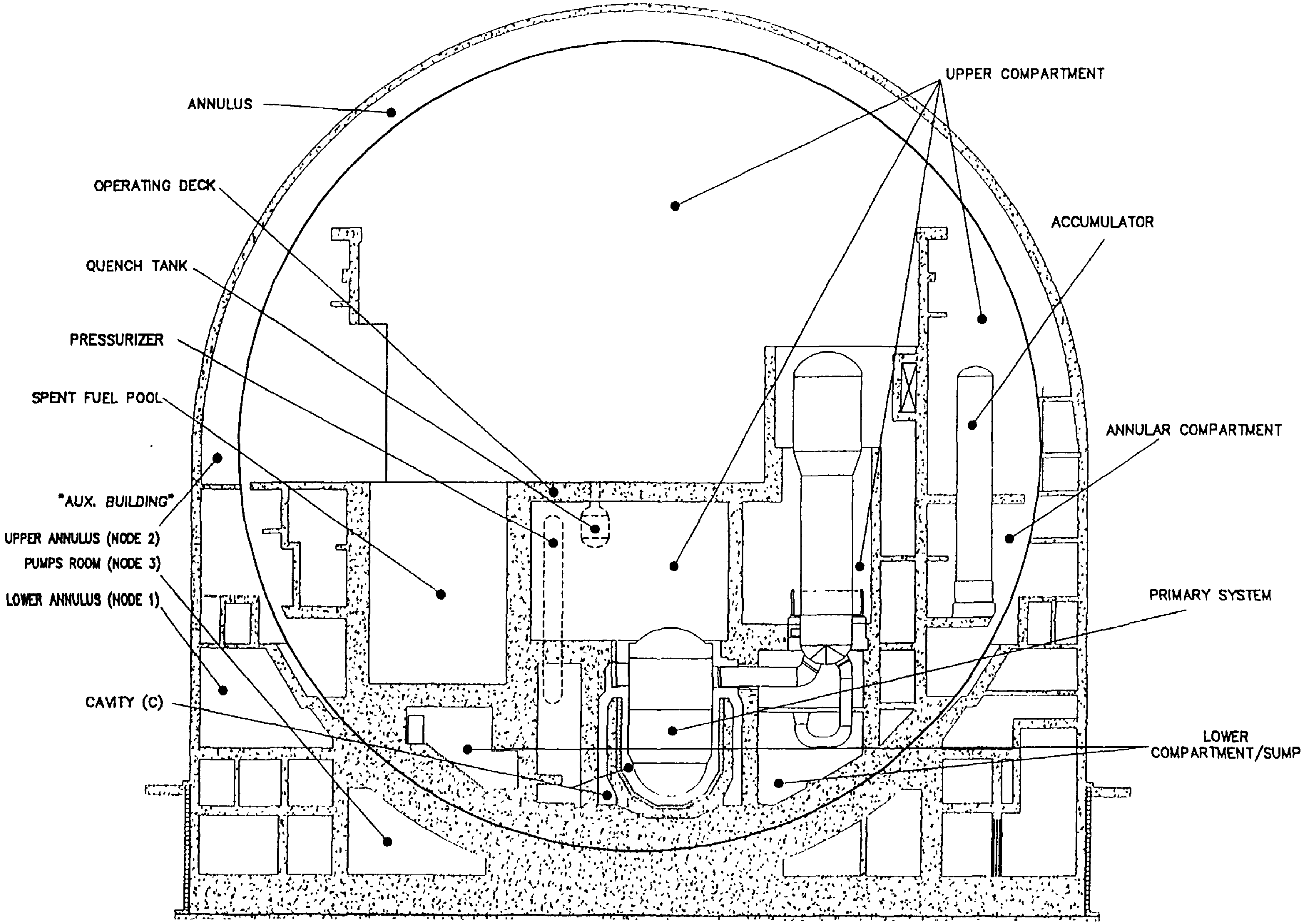


FIG. 3.1 C.N. TRILLO CONTAINMENT

2.3.- PWRs SEVERE ACCIDENT MANAGEMENT

The Spanish PWR of Westinghouse design have taken as their basis for severe accident management the generic SAGs developed by the Westinghouse Owners Group (WOG). Contrarily to the BWROG's approach, in the WOG's the SAGs are clearly separated from the EPGs, and they are only managed by the Technical Support Center (TSC)

To date, the results of the Level 2 analyses available in some of these PWRs show that containment venting is not needed.

3.- TRILLO NPP CONTAINMENT ANALYSIS

C.Trillo-1 is a PWR of KWU design. Its thermal power is 3015 Mw_t. The RCS has 3 loops with 3+1 safeguards trains for the safety injection and the emergency feedwater systems. The design of the safety systems with 4 redundancies stems from the application of the single failure criterion and the consideration of 1 train being repaired.

The containment analysis for Trillo has not been started yet. Currently its Level 1 is being updated with plant modifications implemented recently. However, the Level-1/Level-2 Interface has been analyzed and preliminary Plant Damage States (PDSs) have been defined.

The containment for Trillo, as a KWU design, presents some particularities when compared to more extended PWR designs (see Fig.3.1). It consists on a steel sphere standing on a concrete bowl. This external bowl supports both the sphere and an inner bowl which bears all the containment structures (walls and equipment). The space between the sphere and the concrete containment (the so called "Annular Building") hosts the safety injection system (including its borated water tanks) and the nuclear component cooling system. The steel sphere is designed to stand a LBLOCA in coincidence with a MSLB, so that its design pressure is rather high. The spent fuel pool is located inside containment close to the refuel pool.

The Reactor cavity is dry and tight. For such a tight cavity, the German Risk Study (DRS) assumes that in sequences with high-pressure Vessel failure, containment would fail as a result of the Vessel "rocket" displacement (Ref.1).

Preliminary studies for sequences with medium-pressure Vessel failure show that the fraction of core debris dispersal out of the cavity may be important because of its fluidification and entrainment in such a tight cavity.

Another characteristic is that the containment is partly compartmentalized, although the compartments are rather open. Moreover, there is no spray nor fan-coolers within it for DBA, so that steam inertization is thought to play an important role in Hydrogen phenomena in severe accidents.

If core debris is not frozen, the generation of non-condensible gas due to CCI is very reduced because of the basaltic composition of the basemat concrete. The steady pressurization of the containment in such a situation is rather due to the steaming of water in the sump, because of the decay power of the fission products aerosols deposited in it. However, this pressurization process is very slow. The basemat depth is close to 7 m, so that its melt-through is also a long-term process.

4.- COFRENTES NPP CONTAINMENT ANALYSIS

CN Cofrentes is a BWR/6, with a Mark-III containment, designed by General Electric. Its nominal power is 2951 MW_t. It has three electric divisions, with a safeguard diesel per division. Its commercial operation began on May 1985.

Currently, the Level 2 studies are being performed in Cofrentes. In the Level 1/Level 2 Interface the resulting PDSs have been merged into 16 KPDSs. None of the identified KPDS shows characteristics that could result into large releases in the short term. The most significant KPDS involve transients which ended up in intact containment with both low and high pressure; low pressure critical ATWS with containment failed, and low pressure SBO sequences with intact containment. The Containment Event Trees (CET) and Decomposition Event Trees (DET) are being developed for these KPDSs. Typically, two different CETs are to be used, one for V sequences and the second for the other PDSs.

The containment capability is being determined by a detailed structural analysis on ultimate failure pressure and associated fragility curves. Its dynamic response will be considered only in a simplified way. Key points in the assessment of the containment integrity are:

- capability of the pedestal to withstand large dynamic pressures, high temperatures and corium penetrations up to 8 feet;
- drywell behaviour on underpressures caused by H_2 explosions in the wetwell
- containment quasisteady pressurization.

The containment failure mode is dependent on corium coolability: if corium is not coolable, the pedestal would be challenged by high temperatures and concrete penetration and containment would be pressurized by the non-condensible gases stemming from CCI; if corium is coolable the vapor generation would pressurize the containment, if heat removal is not available. and challenge its integrity.

From the studies developed up to date for Level 2, the main components for containment performance in severe accidents have been identified. These main components, which have an outstanding role in the containment performance of Cofrentes, are the following:

- H_2 igniters. They are located in the containment out of the drywell. There are 2 trains of igniters, one of them ac/dc supplied. They were not installed by design, but rather as an anticipation based on previous American BWR experience;
- containment venting without filtering. It was originally installed as a way to control pressure in emergency conditions (operator action in the EOPs);
- drywell vacuum-breakers. They were installed to avoid loss of coolant in the suppression pool. Their stuck opening would result in a bypass of the suppression pool as a fission product (FP) scrubber;
- containment spray. Even if its heat exchanger were not available, it would have an important effect on FP scrubbing;
- late injection to the RPV or the Pedestal by the Fire Protection pumps.

5.- CONCLUSIONS

The overview of the PSA/IPE studies in Spain has been shown. The plants are following and developing sound and well-known methodology and tools. The studies are well oriented towards, on one hand, the understanding of the key phenomena and plant response in severe accident sequences and, on the other hand. the development of accident management guides.

REFERENCES

1.- NRC 2241, German Risk Study on Nuclear Power Plants, Phase B. Feb. 1990

ANALYSIS OF FEEDWATER ACCIDENTS WITH THE MELCOR CODE FOR THE WWER-440/213 NPP

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Abstract

This paper contains a description of the Level 2 PSA related studies in Hungary for the Paks NPP. A Level 1 PSA has been performed for the Paks Unit 3 in the framework of the AGNES project which is an overall safety reassessment project for this plant. The current PSA topics in work are shutdown PSA and human reliability. Also limited Level 2 PSA studies were performed for the AGNES project. The analytical tools available for the Level 2 work are STCP, CONTAIN, MELCOR, RELAP-SCDAP, and MAAP. Most of these codes are running at VEIKI. Good experience is available for STCP, CONTAIN and MELCOR, and currently experience is being gained for MAAP and RELAP-SCDAP. Still more experience is needed for all the codes.

1. INTRODUCTION

Severe accident studies in Hungary started in VEIKI in 1988 when the Source Term Code Package (STCP) was adapted to VVER-440 reactors with the help of the IAEA (permitted by US NRC). After several initial studies the STCP was used extensively for the AGNES Project [1]. Since the time the AGNES project was completed, a new, much more developed and versatile code, the MELCOR code was adapted for the analysis to eliminate the limitations built in into the STCP. This presentation summarizes the latest results and experience with the MELCOR code.

2. INPUT DEVELOPMENT

The VVER-440 input deck has been installed on the IBM_RISC_6000 workstation of VEIKI. A comprehensive set of plots has been set-up to evaluate the computed results. The input developed by the help of the IAEA (RER/91004) was used as a starting point. The whole input deck has been re-structured and re-assembled and adapted to the MELCOR 1.8.3 version. Steady-state conditions have been obtained by imposing time independent volume conditions in the steam generator (SG) and pressurizer. Steady-state plant parameter values have been re-introduced to the input to obtain steady-state in a shorter time. The steady-state obtained is close to normal operating conditions. The code keeps this obtained steady-state well. The nodalization scheme of the primary and containment side is given in Figs. 1 and 2.

The following problems have been resolved:

Primary circuit flow-path junction opening heights were adjusted to obtain "smooth" level decrease in the reactor vessel (that is avoid level decrease in the core before most of the upper plenum (UP) gets empty). Momentum exchange length parameters have been introduced. Primary loop cold- and hotleg flowpath junction elevations have been changed to model loop seal clearings correctly.

Pressurizer (PRZ) heaters switch-off setpoints on low level in PRZ have been corrected. Also the PRZ pressure operated relief valves (PORV) setpoints have been corrected according to the VEIKI report 21.11.104-2.

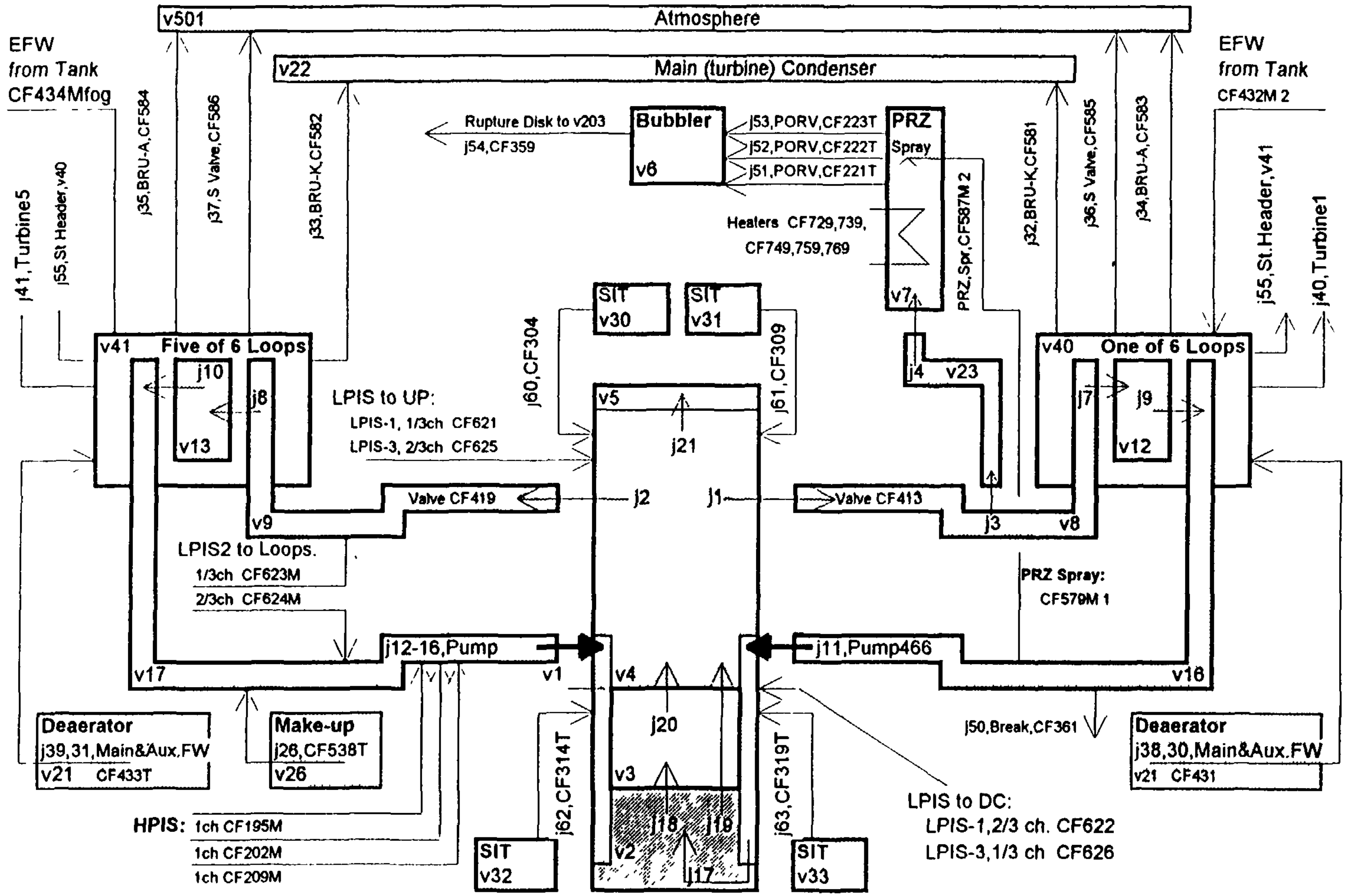

Fig. 1 MELCOR Primary Circuit Nodalization and Flow-Path

Incl. Primary Circ. Cont. Breaks **VVER-440/v213**

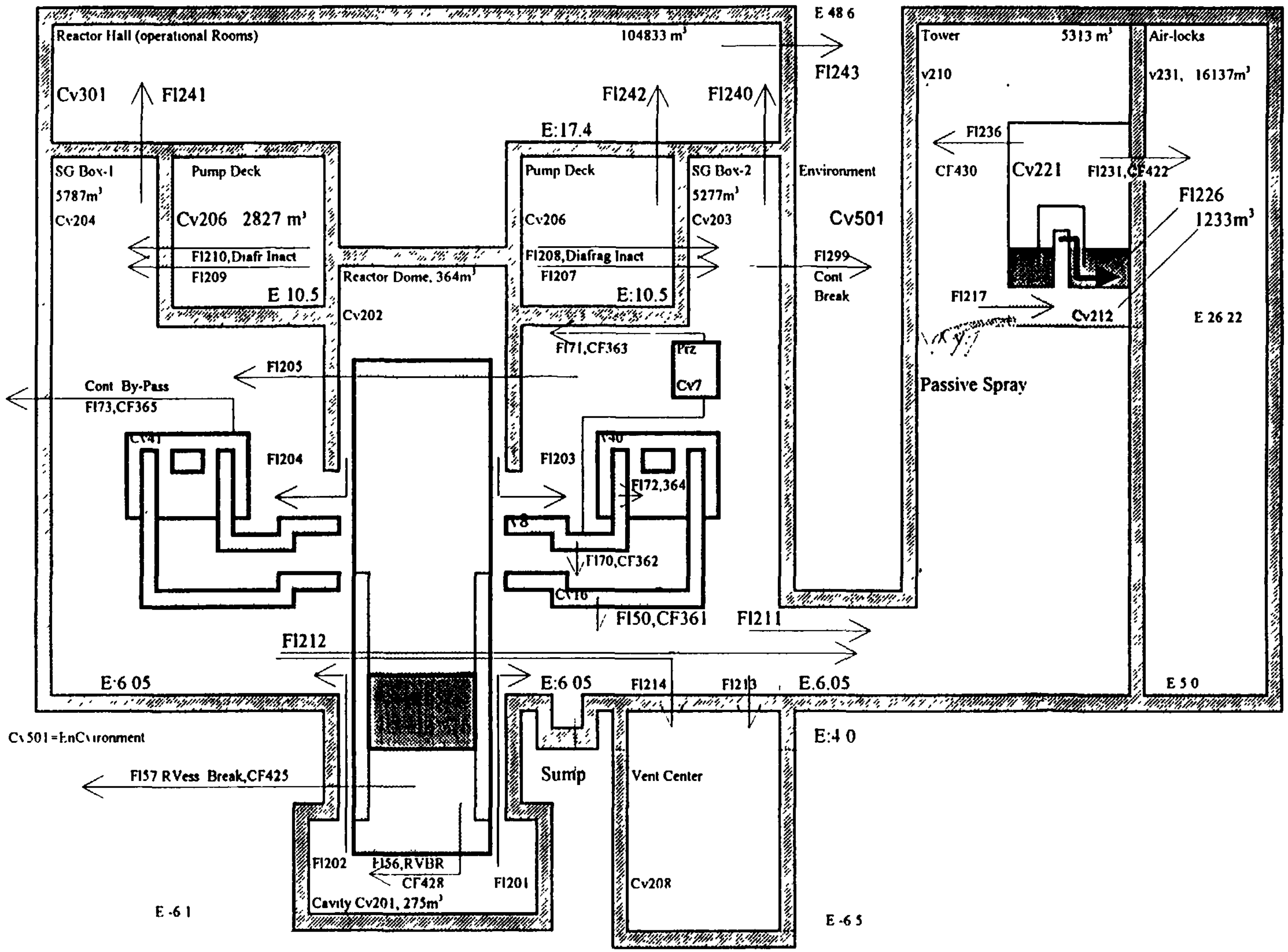

Fig. 2 MELCOR Containment Nodalization and Flow-Paths

Hydro-accumulator's wall to containment has been provided with convective heat transfer boundary conditions (and increased heat transfer coefficient) to avoid temperature condition in their atmosphere being out-of-the-range of validity of MELCOR equations.

Main Coolant Pump (MCP) models have been corrected by introducing a MCP trip signal unit (PUMPTRIP.3IN file) handling the technological signals leading to MCP trip. (Pump trips are driven now only by pressure, loss-of-electric-power and manual trip signals. Certain kind of pump trips lead to SCRAM.)

Primary Circuit Breaks are introduced as valves activated after an initial steady-state run in the restart input of the MELCOR (MELIN) input file.

The **SCRAM SIGNALS** model has been corrected. (Now any SG level decrease below -0.4m results in SCRAM).

The **CORE package** input has been adapted to MELCOR 1.8.3(dT/dz model). Debris surface area of Lower Head (LH) penetrations (CORPEN card Variable:ASPN) has been taken to be a small number to avoid penetrations, not existing in the VVER-440s Lhead.

On Secondary Circuit side new models for Main Feed Water, Auxiliary Feedwater, Emergency Feedwater and Turbine models have been introduced. Heat transfer to secondary side has been solved with 5 separate heat structures along the height of the SGs. This model provides a correlation of the heat transfer with the level in SG on both primary and secondary side.

The **Containment Bubbler Tower Trays** model has been simplified to avoid excessive calculation burden but retain accuracy. (Trays downcomer CV216 has been removed, and a direct horizontal flow-path to trays has been introduced with junction elevations reflecting the water lock opening and the back-flow of water from trays.) The new model has been checked including the check-valves from trays to air-locks and from trays to back to tower.

The **Passive Spray model** from trays of the bubbler condenser has been corrected. (Water supply to Passive Spray source (CV212=space below trays) has been provided as a flow-path.)

The **Cavity** package has been re-activated to calculate core-concrete interaction with simplified geometry.

The **BUR**ning and Radionuclide (**RN**) package has been activated.

In **PRZ spray, Passive Sprays, SG Emergency Feed Water** models instead of previous erroneous version mass and enthalpy sources have been specified.

The High Pressure (BPIS) Low Pressure (LPIS) injection parts of Emergency Core Cooling (ECC) and the Containment Sprays systems have been reviewed, corrected and modified including several new features. The new features are the heat removal calculation in the Spray and ECC heat exchangers and a containment sump model.

3. CALCULATING ACCIDENT SEQUENCES

The sequences calculated started from normal operating conditions (steady-state).

3.1. Loss-of-Coolant Accident (d=100 mm)

A severe nuclear accident initiated by a single ended pipe break of d=100 mm assumed to be in the lowest point of the cold-leg and accompanied with total loss of electric power (including diesel generators) has been analyzed.

Calculation was proceeded up to 7200 minutes (432000 s) without RN package and up to 625 minutes (37500 s) with RN package. Bottom head failure happened at 8340 s.

3.2. Total Loss-of-Feedwater Accident

3.2.1. Sequence definition

Results of the PSA Level-1 studies performed in the frame of the AGNES project [1] showed that in case of the VVER-440/213 nuclear power plant at the PAKS site, one of the biggest contributors to the core damage frequency is the severe nuclear accident initiated by the loss of (main, auxiliary and emergency) feedwater due to instantaneous full cross section break of the main feedwater header which also breaks the emergency feedwater header situated close to the previous one. After the initiating event the reactor scram stops the chain reaction but the loss of secondary heat sink leads to core damage.

The electric power will be available during the whole process but the make-up water (charging water) and the active ECC can not discharge due to the high pressure in the primary system. The accumulators can start only after the vessel failure. The containment spray system is available during the whole process.

It was assumed that the check valves between the SG and main feedwater header are closing very quickly so that no significant secondary coolant loss to containment happens.

3.2.2. Sequence timing

The main events and timings are the following:

- the main, auxiliary and emergency FW stops at time 0.0s;
- loss of FW results in rapid decrease of FW in SGs which causes a SCRAM signal (at 55s) when the level is 0.4 m below the nominal;
- dry-out of the SGs causes deterioration of the heat-exchange in the SGs which in turn starts the primary circuit pressure to rise and initiates the discharge of the primary coolant to the bubbler tank of the pressurizer;
- the membrane of the bubbler tank breaks to the containment at a pressure of 1.52 MPa in the bubbler vessel and the discharge of the primary coolant to the containment starts from 12797 s;
- the Main Coolant pumps stop due to the signal "Containment Pressure > 0.11 MPa" at 13891 s;
- the core heat-up and melting starts when the water level in the reactor vessel decreases below the top of the core at 20000 s;
- the release of fission products from the gas volume of the fuel cladding (gap release) to primary circuit happens during 21124-23378 s;
- failure of the core support plate in the 4 radial rings used happens in the time interval of 28301 - 53184 s;
- hydroaccumulators start to charge water to reactor only after the reactor failure at 91198 s;
- the whole process has been calculated up to 6000 min.

3.2.3. Sequence major findings

The major conclusions are as follows:

- during the high pressure vessel failure a nearly 4 MPa pressure spike develops in the reactor cavity, that may challenge the containment integrity;

- retention of volatile fission products in the primary circuit after the reactor vessel failure is relatively small due to high temperatures in the primary circuit. At the same time heat-up of the hot-leg and its subsequent failure may change the course of events. High pressure vessel failure may be avoided and a loss-of-coolant (LOCA) like sequence may develop. For final conclusions more detailed (structural response type) analysis would be needed;
- for most of the fission product (FP) classes (except Te and Ce classes) release from the fuel **during in-vessel melting** phase is the most dominating;
- release **from primary circuit to containment** is relatively low during the in-vessel melting phase for most of the FP classes. A high intensity source appears at the reactor vessel lower head failure, when the FP vapors and aerosols (floating in the vessel) get released. For FP classes (e.g. Te) where the ex-vessel release is more important (or if the release of volatile FP was not finished during in-vessel phase) another rise in containment FP concentration was observed;
- at the end of calculations (after 6000 min) most of the volatile **Xe, CsI, Cs, Te** classes got released to the containment. The release to the containment of the other FP classes was small. In particular release of the **Ba, Mo, Cd, Sn** classes from the fuel was relatively high but most of this release was retained by the core volume or by the primary circuit. Nuclides in FP classes of **Ru, Ce, La, U** almost entirely remained in the fuel. Only a small fraction of the Ru class nuclides appeared in the primary circuit;
- most of the FP radionuclides at the end of calculations are in the SG-box containing the release place and in the bubbler tower containing the corridor connecting the SG-box to the bubbler tower;
- release of the FPs (except Xe) from containment is smaller than the previously calculated maximum due to the operation of the containment spray system. The release from containment for Cs and CsI is 0.032 and 0.055 % of initial core inventory respectively. A similar release from containment was experienced in case of Te (0.0865 %). The total Cs release (including Cs and CsI classes) is 0.0371%;
- FP distribution after 6000 minutes is given in Table 1. according to MELCOR calculations. FP distribution for the same conditions calculated by STCP are given in Table 2. The largest difference found was the higher release of volatiles from the primary circuit calculated by MELCOR which is attributed to higher temperatures on the primary circuit calculated by MELCOR.

Table 1. FP radionuclide distribution in a VVER-440 plant after 6000 min according to *MELCOR* calculations in percent of initial core inventory for LOFW sequence. (No credit to retention in secondary confinement is given.)

Nuclide Class		In-Fuel %	In Primary Circuit %	In-Containment %	Released from containment %
1	**Xe**	3.42E-08	0.17	72.59	27.24
2	**Cs**	3.79E-07	10.68	89.29	0.032
3	**Ba**	70.60	26.76	2.64	0.0021
5	**Te**	27.07	3.18	69.66	0.0865
6	**Ru**	98.83	1.16	0.0109	2.55E-06
7	**Mo**	83.22	16.60	0.18	5.32E-05
8	**Ce**	99.77	0.0286	0.20	0.0001
9	**La**	99.92	0.063	0.0137292	1.55E-05
10	**U**	99.94	0.0557	0.0081	1.17E-05
11	**Cd**	41.74	56.72	1.54	0.0021
12	**Sn**	40.96	56.78	2.26	0.0036
16	**CsI**	0.35	11.10	88.50	0.055

Table 2. FP radionuclide distribution in a VVER-440 plant after 6000 min according to *STCP* calculations in percent of initial core inventory for a sequence similar to LOFW. (No credit to retention in secondary confinement is given.)

Nuclide Class		In-Fuel %	In Primary Circuit %	In-Containment %	Released from containment %
1	**Xe**	0	0.05	82.35	17.6
2	**Cs**	0	90.11	9.88	0.0067
3	**Ba**	75.21	5.83	18.87	0.07
5	**Te**	26.79	61.93	11.17	0.06
6	**Ru**	100.00	0.0007	0.000096	1.8E-06
7	**Mo**				
8	**Ce**	99.86	0	0.14	0.00054
9	**La**	98,02	0.00014	1.98	9.0E-04
10	**U**				
11	**Cd**				
12	**Sn**				
16	**CsI**	0	86.87	12.12	0.0074

Abbreviations

ECC	Emergency Core Cooling System
LPIS	Low Pressure Injection System
LOFW	Loss of Feedwater (accident)
HPIS	High Pressure Injection System
SG	Steam Generator
FP	Fission Product
PRZ	Pressurizer
HA	Hydroaccumulator
UP	Upper Plenum
LH	Lower Head
AGNES	Advanced General and New Evaluation of Safety (of the Paks NPP)

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[1] AGNES PROJECT: Safety Assessment of the PAKS Nuclear Power Plant, Executive Summary, Oct.1994, KFKI, Budapest, Hungary

XA9846921

OVERVIEW AND CURRENT STATE OF THE LEVEL 2 PSA FOR THE BALAKOVO WWER-1000: ANALYSIS OF CONTAINMENT PERFORMANCE

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Abstract

This paper summarizes the current state of the Level 2 PSA work for the Balakovo Unit 4 NPP. First, the Level 1, 2 PSA interface was developed and represented by a set of interfacing event trees. The resulting plant damage states (PDSs) were used as input to the containment event tree (CET) developed for the Level 2 PSA. Consequences of the CET are grouped into radiological release categories (RCs). Allocation of RCs in the CET is based on the similarity of the sequence characteristics. Each RC may be quantified in terms of the potential fractions of core inventory of radioactive material that may be released to the environment and also by the characteristic of releases such as time to release, release duration and availability of warning time. Containment capability analysis has been performed with the ABAQUS code which shows a high level of the containment ultimate pressure. The containment fragility curve was developed taking into account major randomness and uncertainties sources in containment characteristics. Analysis of containment leakage and of the performance of penetrations was also performed. As a result, the containment failure mode under internal pressure rise was defined as a global failure of the containment in the membrane area of the cylinder. As the data on severe accident analysis was only available for large LOCA, the quantification of CET sequences was limited to one sequence. The nodal probabilities were introduced based on data from the analysis of the large LOCA, containment fragility curve and expert judgement. The most probable sequence is associated with low pressure failure of the RPV and basemat meltthrough through instrumentation channels with further melting of the foundation slab and release to the environment.

1. INTRODUCTION

Under the TACIS 91 Project 3.1 PSA Level 1 and 2 including analysis of containment performance and seismic PSA are carried out for the Balakovo NPP, Unit 4. Balakovo Unit 4 is a WWER 1000/320 plant. The purpose of the project is the transfer of up-to-date methodologies, and provision of software and hardware to perform the PSA. The following organizations participate in the project:

- Atomenergoproekt
- Hydropress
- Kurchatov Institute
- Balakovo NPP
- NNC Ltd., UK
- Empresarios Agrupados, Spain
- Belgatom, Belgium
- AEA Technology, UK

2. LEVEL 2 PSA INPUT DATA

The Level 1 PSA results are used for the definition of plant damage states (PDSs) for the Balakovo NPP. The PDSs represent the input for the containment event tree (CET). The PSA Level 1 - Level 2 interface is represented by a set of interfacing event trees used for quantification of the PDSs.

Further information is available from the severe accident analysis which was carried out under the TACIS 91 project 3.8. However, results are only available for the Large LOCA which was the basic constraint of this analysis.

The Balakovo WWER-1000 containment capability evaluation, which is described in more details below, considered the ultimate containment behavior under internal pressure. This analysis revealed the containment failure modes and the containment fragility curves for each mode were determined.

3. CONTAINMENT EVENT TREE DEVELOPMENT AND QUANTIFICATION

Fig. 1 shows the containment event tree developed for the Balakovo NPP. The following phenomena are not explicitly modelled in the CETs:

- Early containment failures caused by steam explosions (α - mode failures)
- high pressure melt ejection (HPME)
- direct containment heating (DCH).

These phenomena are excluded from explicit modeling in the CET on the basis of recent studies for PWR reactors and because of specific design features of the WWER-1000 reactors.

The following results from severe accident and containment capability analysis for the WWER containment are important from the point of view of the CETs:

- The amount of hydrogen generated within the containment is unlikely to cause significant hydrogen burning affecting containment integrity.

- Basemat failure and consequent release of radioactive material through instrumentation channels to non hermetic rooms seems to be very likely, see Fig. 2 showing the layout of the reactor cavity.

- There is no information available to assess whether the foundation slab integrity can be maintained. However, such event is estimated as "probable" and expressed by a probability value of 0.5.

The endpoints of the CET are categorized into radiological release categories (RC), see Fig. 3. The allocation of release categories is based on similarity of sequence characteristics.

The most probable path from the CET for large LOCA accident sequences is low pressure meltthrough of reactor vessel, no early (and late) containment overpressure, meltthrough of the basemat through instrumentation channels followed by melting of the foundation slab.

4. CONTAINMENT CAPABILITY ANALYSIS

Table 1 shows the main characteristics of the Balakovo WWER-1000 containment. The basic assumption for the analysis is a smooth pressure rise up to the ultimate value. A preliminary analysis was carried out with simple hand calculations using membrane theory to get an estimation for the qualitative assessment of containment behavior. Furthermore a deterministic check of boron sump walls ability to withstand high pressure has been done.

The detailed analysis included the following steps:

- An axisymmetric finite element model of the containment was elaborated using nonlinear material behavior, see Fig. 4 illustrating the finite element model of the containment. The model includes the containment vessel with tendons and reinforcement, and basemat and liner. Large openings are omitted. The purpose was to verify and refine results of the preliminary analysis for prestressed concrete and liner with respect to beyond elastic limits behavior.

  The ABAQUS finite element program is available for modelling of concrete. The result of the analysis is the ultimate pressure of global containment failure.

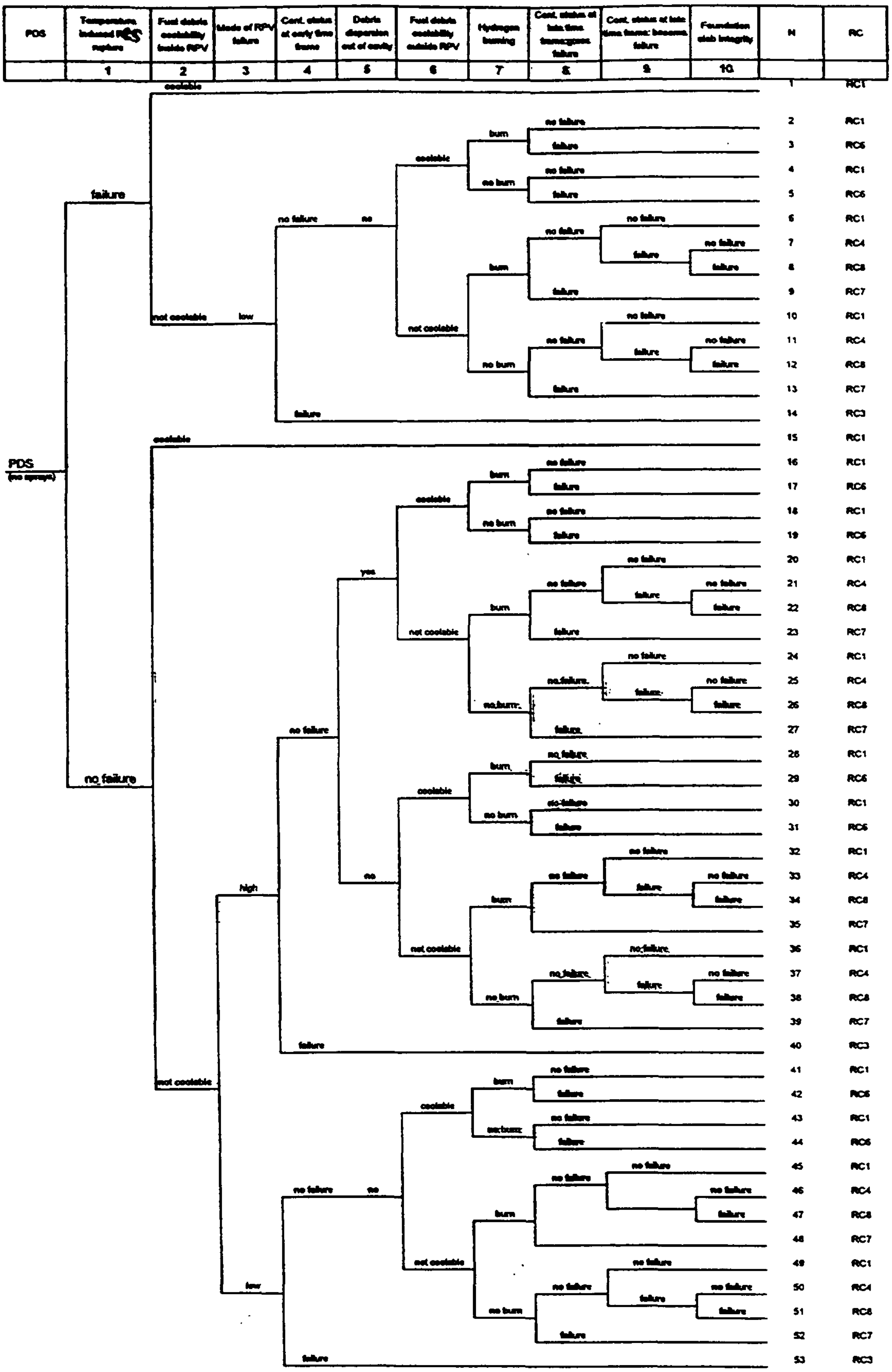


Fig. 1. Containment Event Tree Developed for the Balakovo WWER-1000 NPP.

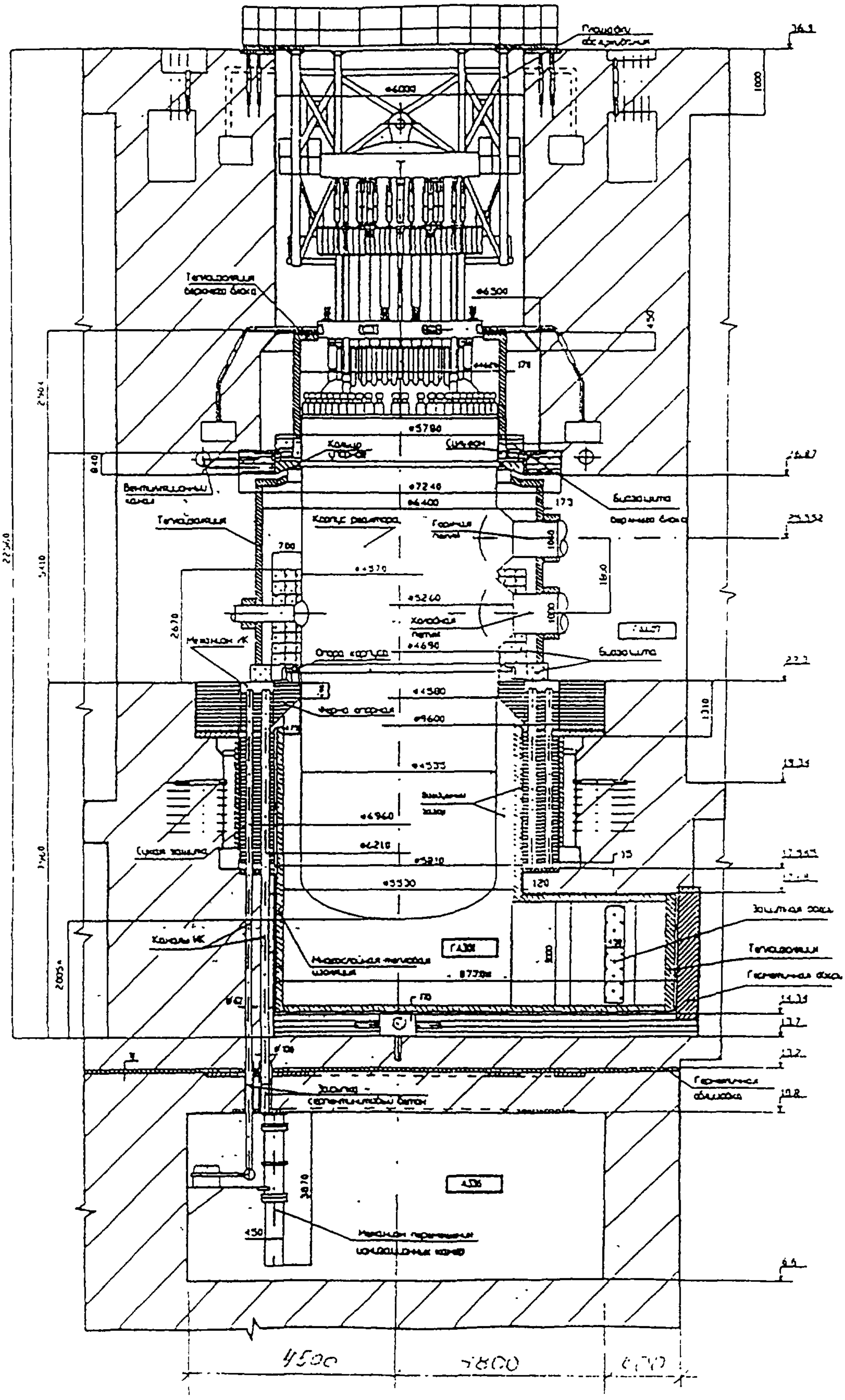


Fig. 2. WWER-1000 Reactor Geometry.

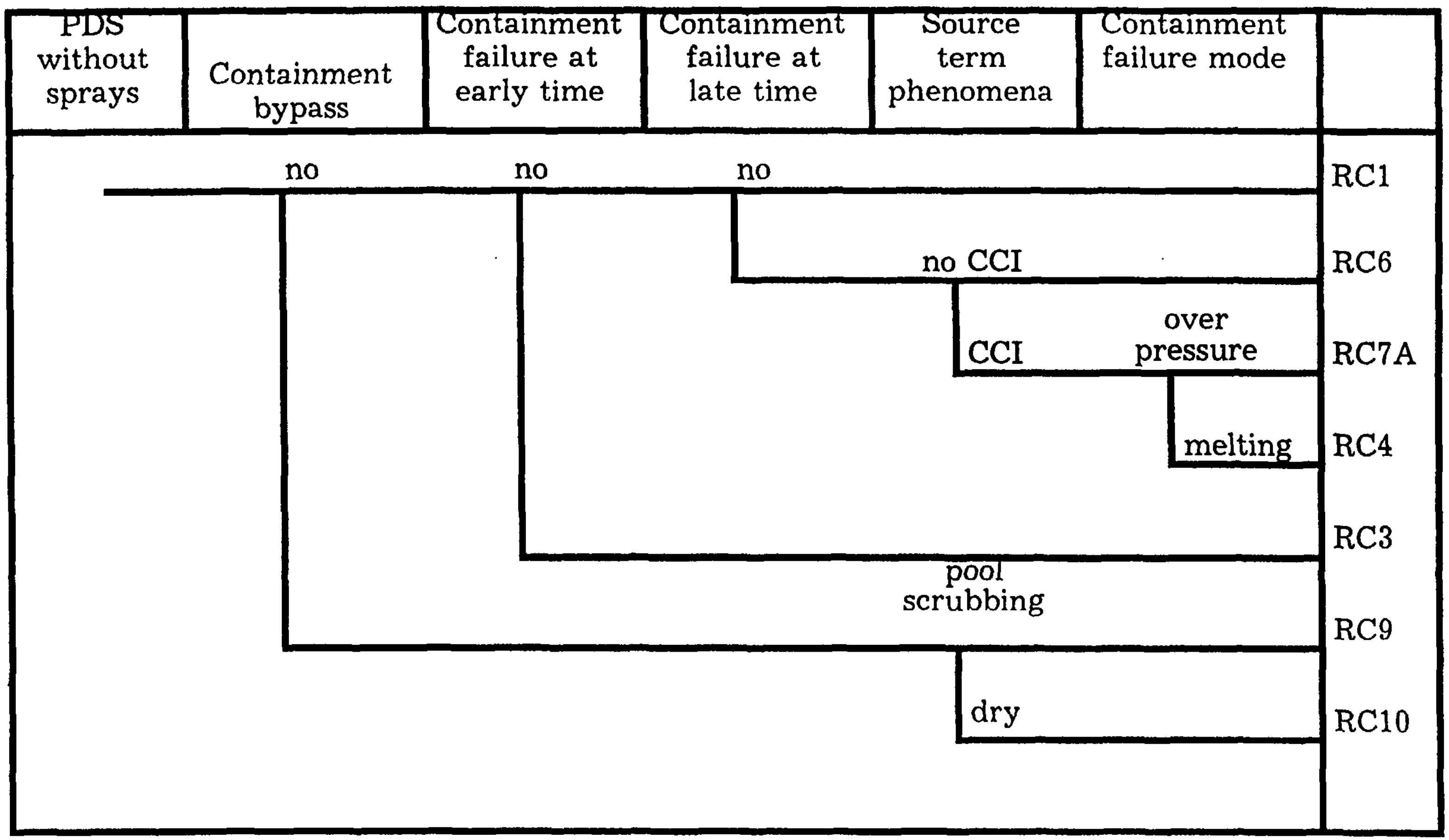


Fig. 3. Allocation of Release Categories.

- Leakage analysis. Analysis is performed for basic strain concentration locations - large penetrations, airlocks, springline, wall-basemat juncture anchorage - using the methodology described in EPRI report "Criteria and guidelines for predicting of concrete containment leakage". The methodology is based on generic data on strain concentrations and takes into account comprehensive test data validated by finite element analyses using ABAQUS.

- 3-D model of equipment hatch in the basemat. The purpose is to obtain strain distribution around this discontinuity in order to assess the potential for leakage.

- Penetration capability analysis. Analysis is aimed to check that penetration components such as insert plates, gaskets and welds are able to sustain the ultimate containment pressure.

Table 1. Main Characteristics of the Balakovo VVER-1000 Containment

Type	Prestressed concrete vessel
Cylinder inner radius Cylinder wall thickness Dome (spherical) inner radius Dome wall thickness	22.5 m 1.2 m 35.0 m 1.0 m
Containment volume Containment design pressure	60630 m^3 0.46 MPa
Basemat	Reinforced concrete 2.4 m
Liner	8 mm steel, anchored (angle & bended rods) to the inner surface of the containment shell
Containment spray	yes
Containment pressure boundary	Liner of dome and cylinder, insert plates at penetrations, equipment hatch and airlocks, liner at the basemat and double liner of the boron solution sump.
Prestressing system	Helicoidally arranged 96 tendons in polyethylene ducts. The tension in the tendons is provided by jacks located in buttress (support ring between cylinder and dome)

5. CONTAINMENT FAILURE PROBABILITY ANALYSIS UNDER INTERNAL PRESSURE RISE

Deterministic containment capability analysis has been performed using median values for the material strength parameters. The resulting ultimate pressure P_m is assumed to be the median value. The random value of the ultimate containment pressure is assumed to be lognormal:

$$\tilde{P}_u = P_m \tilde{\varepsilon}$$

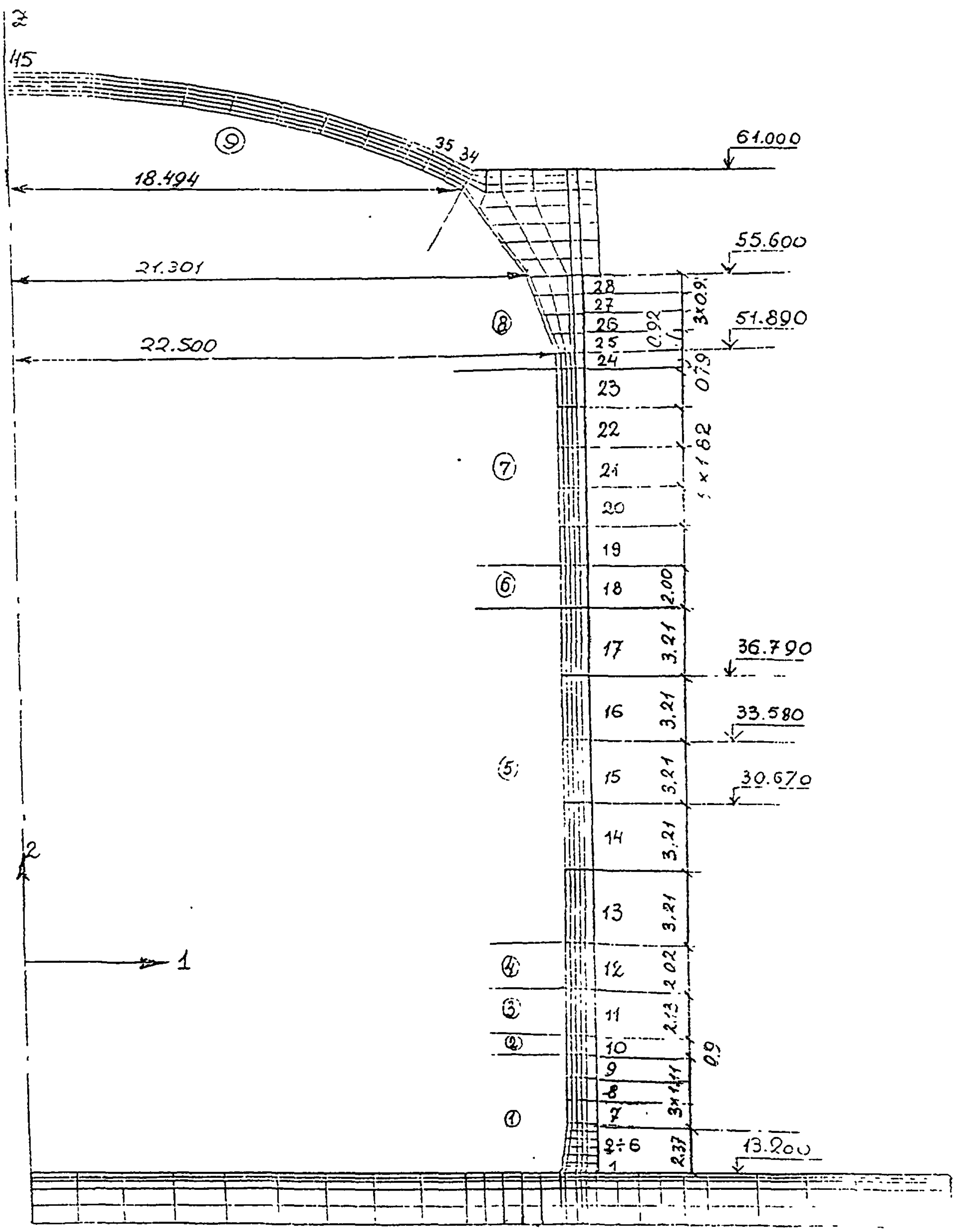


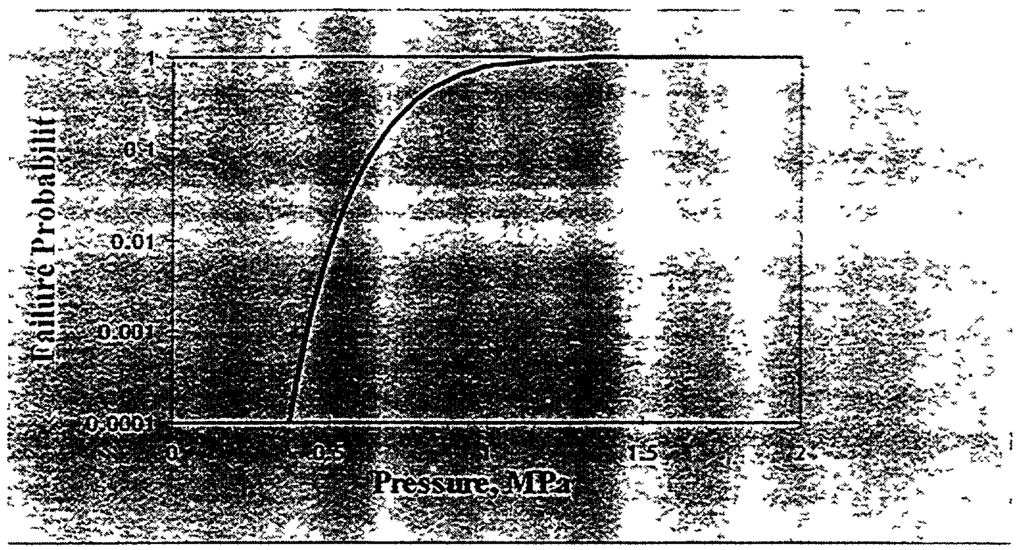
Fig. 4. Finite Element Model of the Containment.

$\tilde{\varepsilon}$ - lognormal random value with unit median which takes into account the following influences:

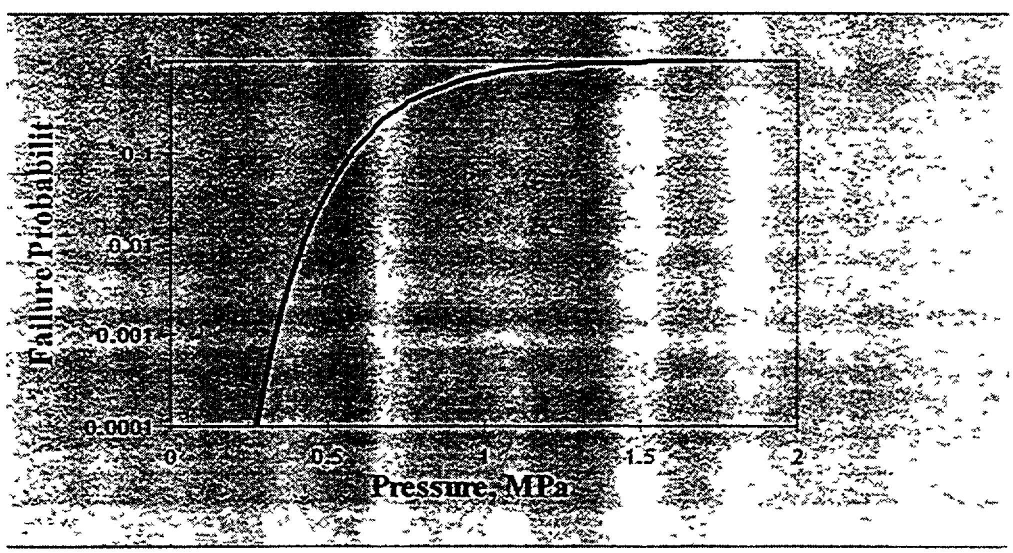
- Discrepancy between the numerical model and the actual structure
- Approximations and small errors in the project and drawings

- Approximations and small errors on site
- Approximations about the behavior of the structure during its time life (settlements) and other uncertainties
- Uncertainties with regard to material (yield and ultimate strength of tendons, rebars, liner plates, concrete strength)
- For leakage analysis - uncertainties with respect to the application of the EPRI methodology to the WWER containment.

The studies show that leakage in the area of the main steam penetration is strongly correlated with global containment failure, which is the result of the rupture of prestressing tendons at the mid-height of the containment cylinder Fig 5 shows the containment fragility curves for global failure and for leakage near the main steam penetration



Balakovo VVER-1000 containment fragility curve for global failure



Balakovo VVER-1000 containment fragility curve for leakage near main steam penetration

Fig 5 Containment Fragility Curves for Global Failure and for Leakage near the Main Steam Penetration

XA9846922

STRATEGIES FOR OPERATION OF CONTAINMENT RELATED ESFs IN MANAGING ACTIVITY RELEASE TO THE ENVIRONMENT DURING ACCIDENT CONDITIONS

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Abstract

In Indian PHWR design, a double containment concept with passive vapour suppression pool (to limit peak pressure) system has been adopted. In addition to it, various Engineered Safety Features (ESFs) have been incorporated to limit the release of radioactivity to the environment. They are:

i) *Reactor building emergency coolers for cooling which results in fast reduction of overpressure;*

ii) *Primary Containment Filtration and Pump Back System (PCFPBS) for reduction in iodine concentration inside RB atmosphere during post LOCA period and*

iii) *Primary Containment Controlled Discharge System (PCCDS) for the rapid reduction of over-pressure tail.*

Due to operation of secondary containment purge system, which maintain negative pressure in the annulus, the ground level release is negligibly small. However, if non-availability of negative pressure in secondary containment space is assumed, then operation of PCFPBS and PCCDS system reduces the ground level release significantly. In this situation, depending upon time of operation of the PCFPBS, it can effectively reduce the iodine release, both in stack level and ground level by trapping it in charcoal filters. It is seen that delay time of PCFPBS operation in conjunction with prevailing weather condition can be manipulated to reduce the effect of stack level release of iodine.

In this paper the containment related ESFs used in Indian PHWR is discussed in brief and the effectiveness of operator actions and management strategies in actuation of the ESFs in reducing the activity release to environment (during postulated accident conditions) will be brought out.

1. INTRODUCTION

In Indian PHWR, the design of containment system has undergone a evolutionary change from single containment dousing system in RAPS to present double containment, suppression pool system with additional associated ESFs. These progressive improvements are aimed to reduce the release of radioactivity following an accident, so that estimated risk can be minimized. While some of the ESFs are designed to operate automatically, others are deliberately kept manual so that proper operator management can be adopted to use these systems more effectively. In the following sections, a brief description of the containment system along with associated ESFs is given and the results of analyses showing the effectiveness of different ESFs have been brought out. Finally, ESFs management strategies in effective operator actions have been discussed.

2. SYSTEM DESCRIPTION

Containment system is briefly described in the following subsections to the extent it is required for subsequent discussion.

2.1. Containment Structure

Present standardized design of containment structure adapts a double containment philosophy, a primary containment completely surrounded by a secondary containment with common raft. Primary containment (PC) is made of prestressed cement concrete (PCC) cylindrical structure capped with a dome of same material. The secondary containment (SC) is made of reinforced cement concrete (RCC) with similar cylindrical structure with dome at top.

To limit the peak pressure arising out of postulated break in primary or secondary high enthalpy high pressure heat transport system, a vapour suppression pool system has been incorporated. For this purpose primary containment volume has been divided into two accident based volume V1 (drywell) and volume V2 (wetwell). Volume V1 contains the high enthalpy, high pressure fluid systems and separated out from the remaining volume called V2. Volume V1 is connected with Volume V2, via ventshaft/distribution header through suppression pool water in the basement. In case of postulated accident when volume V1 gets pressurized, steam and air mixture passes through the suppression pool, where steam gets condensed and cooled air enters volume V2. The design of containment isolation penetrations including airlock, ventilation ducts and other air handling system are such that the philosophy of double containment extends over them. Automatic isolation of containment would be initiated in the event of (i) Pressure rise or (ii) Activity build up in the containment.

A typical pressure transient following loss of coolant accident (LOCA - a double ended reactor inlet header break in PHT circuit) is shown in Fig.1. The containment is designed to withstand the pressure resulting from LOCA or steam line break (SLB).

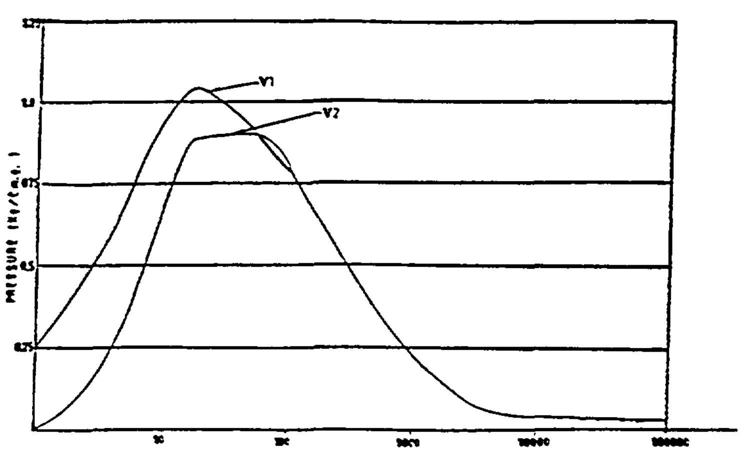

FIG. 1. Cont. Pressure transient.

2.2. Reactor Building (RB) Coolers

The high capacity RB coolers have been engineered to cooldown and thereby depressurise the containment following an accident in a short time, so that the integrated leakage from PC is minimized. They are qualified for functioning under LOCA/SLB environment. These coolers are supplied from assured water and electric supplies. Part of these coolers are normally operating during reactor operation and the remaining ones come on sensing increase in RB pressure or temperature.

2.3. Secondary Containment Recirculation and Purge System (SCRPS)

SCRP system provides multi-pass recirculation via redundant HEPA and iodine filters, within secondary containment space. By purging to stack after filtration, negative pressure is maintained in the secondary containment. While the recirculation provides long term clean up of the secondary containment space following DBA, the provision of purge and resulting negative pressure in the space brings the net ground level release of activity to a significantly low level. This system starts automatically following an accident on high RB pressure/activity signal. The system is also qualified for operation in LOCA/SLB environment.

2.4. Primary Containment Filtration and Pump Back System (PCFPBS)

This system is designed to perform containment atmosphere clean up operation after accident by way of trapping iodine in filter, so that long term release can be controlled to a lower value. This system consists of two identical loops. Contaminated air and steam mixture is taken from volume V1 (by fan induced flow) and passes through a demister (for moisture separation) and a combined HEPA and activated charcoal filter (for particle and iodine removal) and discharged into volume V2. The system is LOCA/SLB environment qualified and fans are provided with assured power supply. Operation of this system is manual by way of operator action.

2.5. Primary Containment Controlled Discharge System (PCCDS)

In order to achieve depressurization at low pressure (say below 0.04 Kg/sq.cm), which may be difficult to achieve by cooling alone, provision is made for restoring to the option of control gas discharge to stack via filters. Also in case, if it is found that the ground level release is taking place as a result of degradation of containment system (i.e. excessive leakage from PC, non-functional SCRPS etc.) or, due to in leakage of compressed air in the RB, the PCCDS operation may be demanded for faster depressurization of PC. This system releases air from V2 space to atmosphere through stack. The flow is controlled by regulating the damper position.

To reduce the compressed air leakage into RB, provision is made to isolate all services air & mask air lines during LOCA. Also supply of instrument air is restricted to only instrument & selected valves, operation of which are essential during the postulated LOCA. Accordingly instrument air supply are made separate, so that non essential supply can be cut off. Such isolation provision has reduced the possibility of air in leakage to a minimum.

Fig.2 shows a schematic of RB containment along with the associated ESFs.

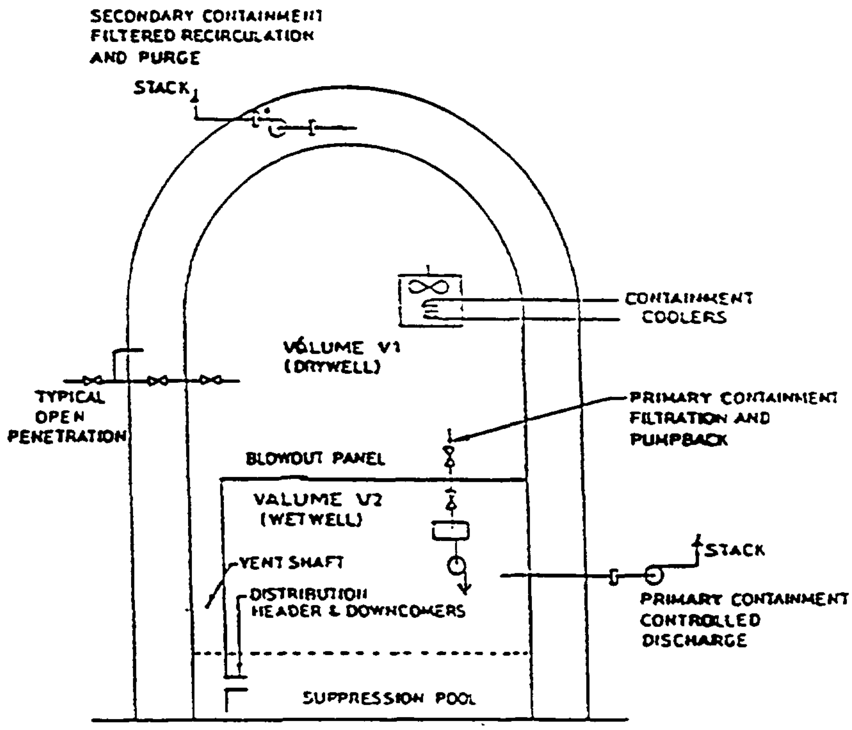

FIG. 2. Schematic of containment and associated safety features in a standardized design.

3. ANALYSIS

Different case studies involving operation delay time of the PCFPB, PCCD & SCRP systems have been carried out to see the effect on activity releases (Iodine & Noble Gases) at ground level and stack level. Also, analyses have been carried out to see the effectiveness of RB coolers in faster depressurization of RB pressure and thus its effect on activity release.

For carrying out the comparative study of ESFs effectiveness, the postulated initiating event considered is a large break LOCA, with conservative assumption with regard to release of fission products (FPs) into the containment atmosphere. Activity release calculations have been carried out for a period of three days following an accident and releases are shown in terms of % release of the initial inventory for all the cases. To have a comparative study, effectiveness of each system is analyzed changing the operation time delay or system capacity with respect to a reference case which considers availability of 50% of reactor buildings coolers, 0.5 hr. delay in SCRPS, 5.0 hr. delay in PCFPBS and 48 hr. delay for PCCDS operation. The result of this case study (ground level and stack level release of Iodine and Noble gases) are shown in Fig.3(a&b). Results of the analyses for different cases are discussed in the following subsections.

3.1. SCRPS Operation

To see the impact of time delay in operation of secondary containment purge and recirculation system, a parametric study has been done considering different time delays from 0.0 hr. to 3 hrs. While the ground level release of Iodine & Noble Gas increases

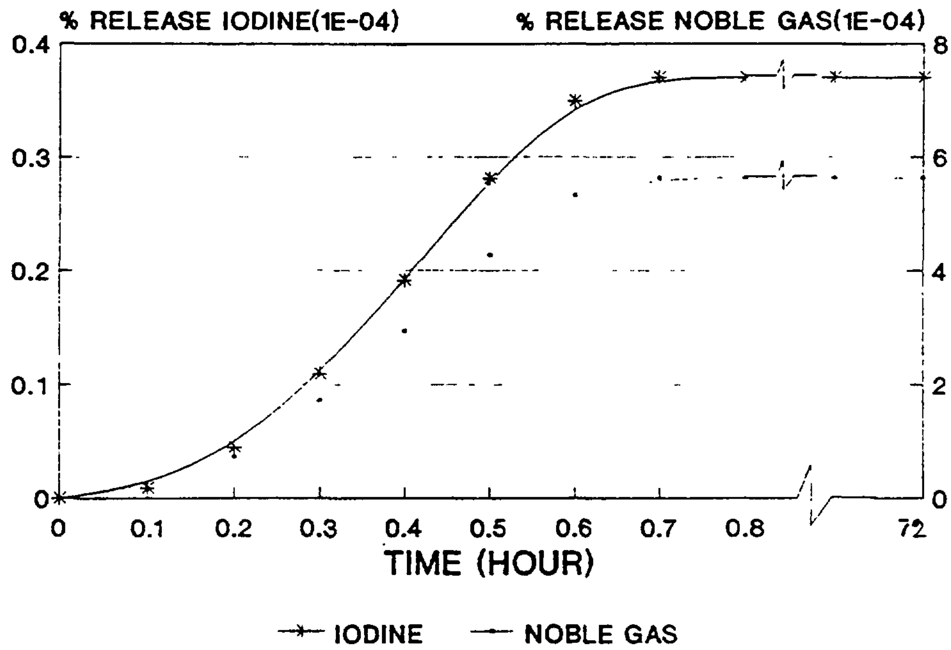

FIG. 3(a). Ground level release vs. Time.

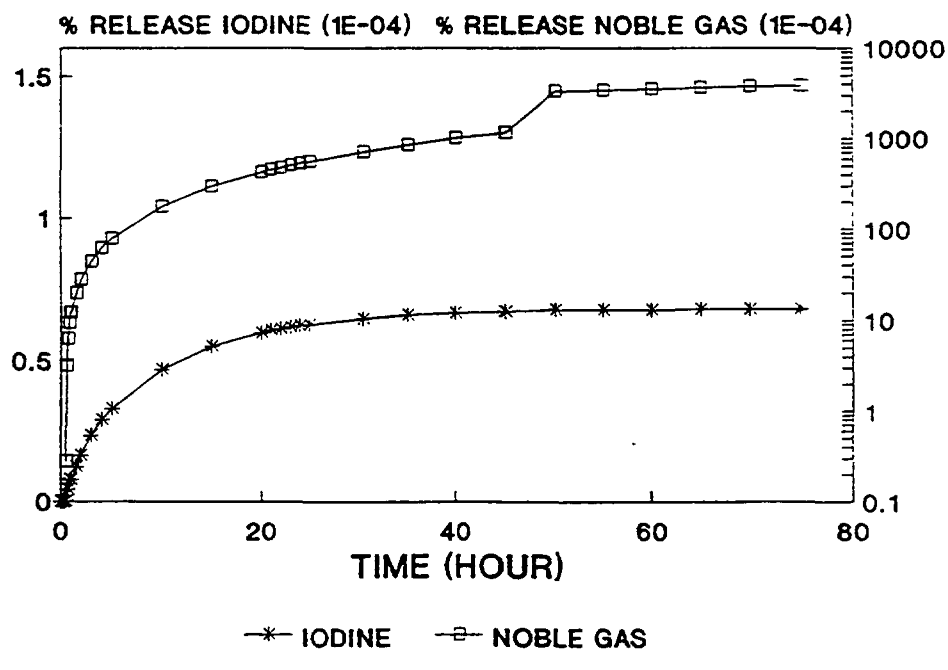

FIG. 3(b). Stack release vs. Time.

with time delay of operation, the stack level release get reduced, since with more delay, releases during the initial period do not exist. While calculating the releases, the purge flow was adjusted to maintain a negative pressure of 12 mm WG in the secondary annular interspace. Fig.4 shows ground level releases of Iodine & Noble Gases with operation time delay. Fig.5 shows the stack releases for the same.

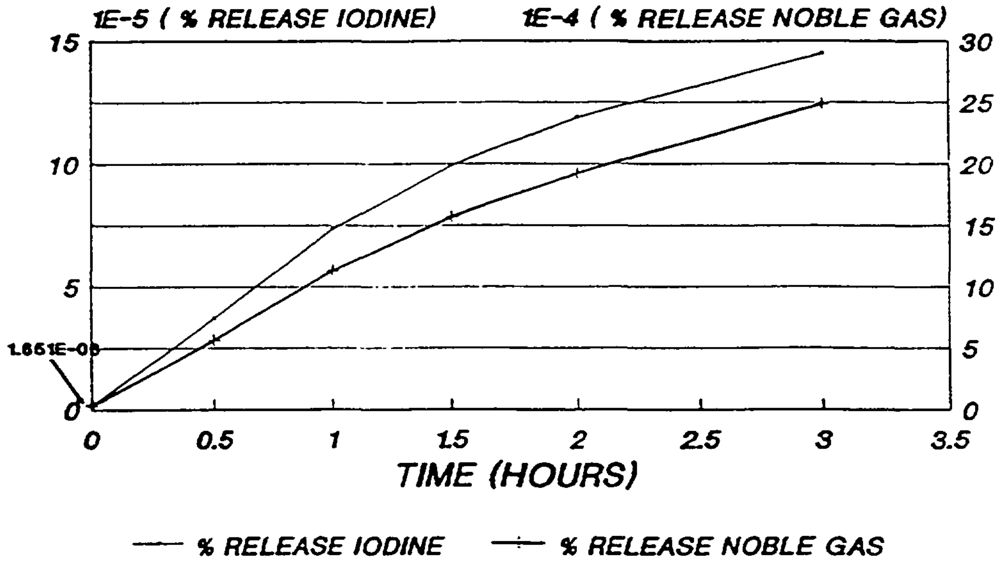

FIG. 4. Ground level release vs. SCRP operation delay time.

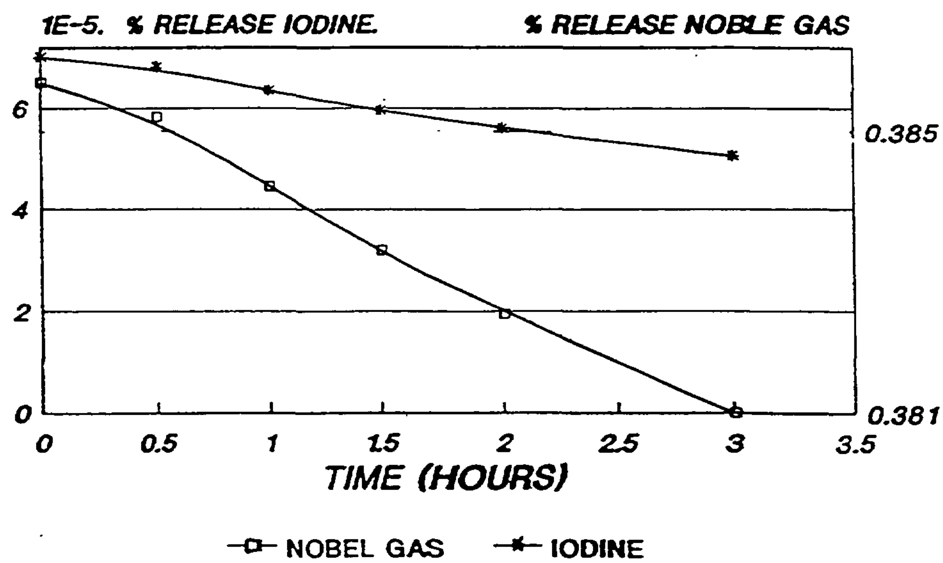

FIG. 5. Stack release vs. SCRP operation delay time.

It is seen from Fig.4 that effect of delay is more during the initial period, because of higher primary containment pressure prevailing at that time, it results in more leakage to secondary containment and thus from secondary containment to atmosphere. The ground level release increase to about 90 times from zero delay to 3 hr. delay period. It is to be noted that stack level release reduces as the delay is more, however, the difference is small. It is thus clear that earliest start of this system gives most effective results.

3.2. PCFPBS Operation

PCFPB system is effective in removing Iodine, but has no effect on noble gas release. Activity release calculation for a spectrum of operating time delays (0.0 hr. to

9.0 hrs.) reveals that main impact of PCFPBS operation is on stack release. This is so, because PCCDS operation results in direct release from primary containment to atmosphere through stack. In case of requirement of early PCCDS operation, the stack level iodine release gets reduced significantly if PCFPB system is valved in early. Iodine ground level release depends on its concentration in secondary containment, which in turn depends on leakage from primary to secondary and the time till the negative pressure is developed in secondary. Since most of the leakage from primary to secondary takes place within an hour, and PCFPBS removes iodine on gradual basis (with effective half life ~2.7hrs.), there will be no significant impact on ground level release of Iodine (Fig.6). However, problem may arise with early operation of PCFPB as a result of choking in HEPA filter & more iodine loading in charcoal filter. Since, at the initial period aerosol and iodine concentration will be high. It is seen from analyses that iodine loading is about 35% more in case of half an hour delay, than that of 9hr. However this factor depends on the filtration flow of the system. Present calculation is based on a designed flow of 13600 m^3/hr.

Fig.6 shows the operating time delay of PCFPBS and consequent release of Iodine. The release through stack is due to secondary containment purge system and subsequent release due to control discharge from primary. In case of compressed air leakage into the containment, periodic release through control discharge system may be required.

3.3. PCCDS Operation

Because of current design of double containment, it is possible to allow the PC to remain at a small overpressure, for an extended period of time following DBA, without adding significantly to ground level release. Thus the controlled gas discharge can be delayed by which time the PCFPB system can remove iodine from the containment atmosphere. With secondary purge system operating, PCCDS operation is called for only when there is pressure build-up in PC during post LOCA period, due to compressed air leakage. In such cases, timing of PCCDS operation affects the stack level

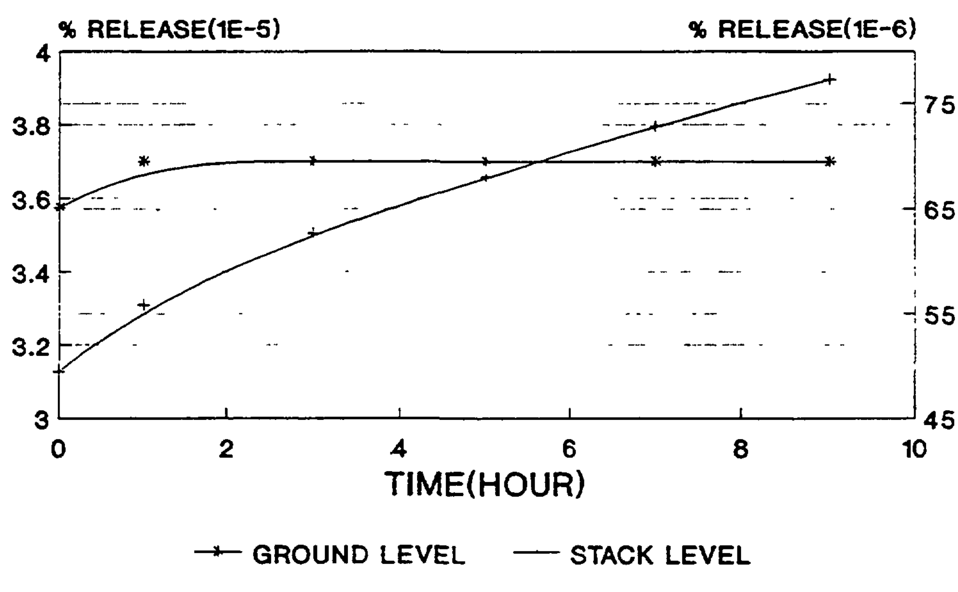

FIG. 6. Iodine release vs. PCFPB operating delay time.

release. Fig.7 shows the stack release of Iodine & Noble Gas vs actuation time of PCCDS. It is seen that stack level release of noble gas continuously decreases with delay of PCCDS operation, mainly as a result of natural decay. Also in case of Iodine, the release decreases even more significantly with delay time and later on stabilizes. This is because of PCFPB operation, which removes the iodine significantly with time (beyond 36.0 hrs.).

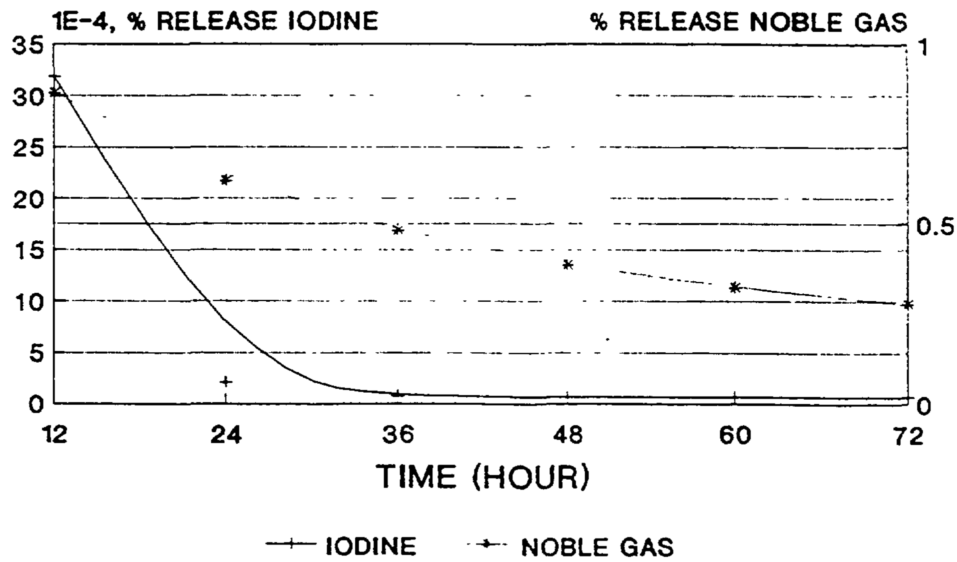

FIG. 7. Stack release vs. PCCD operating time delay.

3.4. RB Coolers

RB coolers help in faster depressurization of overpressure in the containment, which in turn affect the activity release due to leakage from the containment. So as the cooling capacity increases, the release get reduced. This is reflected in the results of analyses given in the following Table. Three cases have been analyzed viz. no cooler (i.e. cooling is only due to natural cooling by way of absorption of heat by containment structures), 50% of total coolers capacity and 100% of the same.

Coolers availability in % of full capacity	Activity Release(%)			
	Ground Level		Stack Level	
	Iodine	Noble Gas	Iodine	Noble Gas
No Coolers	6.5×10^{-5}	9.87×10^{-4}	1.36×10^{-4}	0.577
50% Coolers	3.7×10^{-5}	5.63×10^{-4}	6.8×10^{-5}	0.385
100% Coolers	3.1×10^{-5}	4.67×10^{-4}	6.13×10^{-5}	0.361

It is seen from the analyses results that from no coolers to 100% coolers, activity release can vary by a factor of 2.

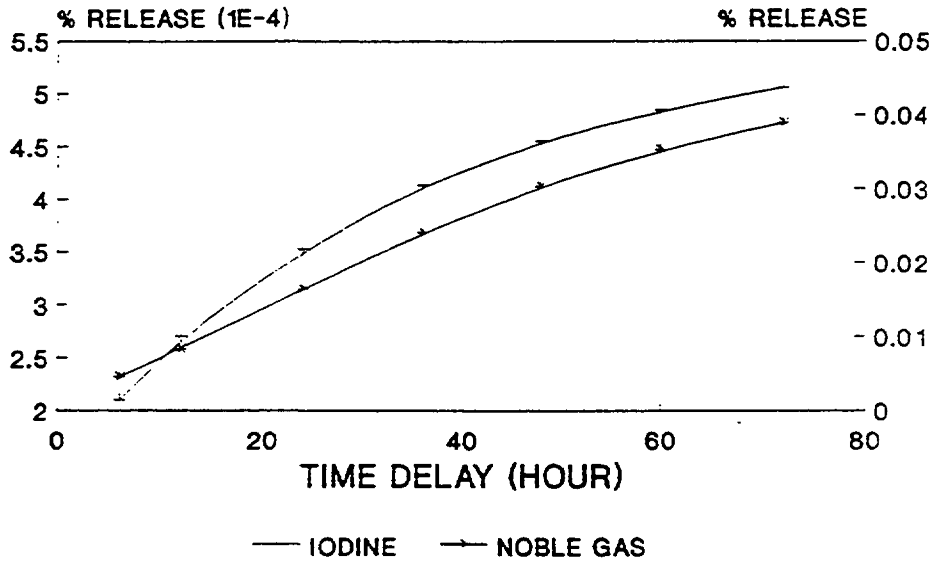

FIG. 8(a). Ground level release vs. PCCD operating time delay.

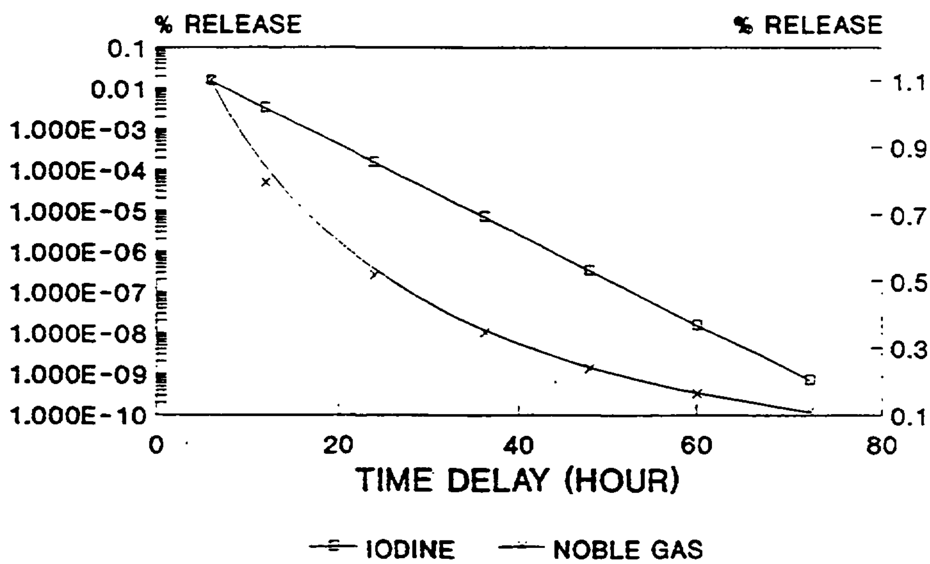

FIG. 8(b). Stack level release vs. PCCD operating time delay.

4. OPERATOR ACTION & MANAGEMENT STRATEGIES

In the above sections the role and effectiveness of different ESFs have been discussed along with results of analyses. Out of four ESFs discussed, operation of RB coolers and secondary purge system is automatic, but PCCDS & PCFPBS operation is manual. In case of accident condition, proper operator action can reduce the release of activity significantly. The required observation and action on the part of operators, in operating the ESFs have been described in the following steps.

4.1. Observations by Operators

Required observation/action for effective use of ESFs are as follows:

i) Assess the level of activity in the RB atmosphere from signals of activity monitors.

ii) Check the condition of compressed air leakage (i.e. rate of leakage) into the containment.

iii) See the status of RB coolers operation i.e. whether there is any partial failure.

iv) Note, whether SC purge system is operating. If in operation, what is the pressure in secondary containment and whether negative pressure is developed in the secondary containment. Negative pressure should be developed quickly (within pre-estimated time), if SCRPS operates as per design intent.

v) Monitor, the prevailing weather condition at the site.

4.2. Management Strategies

From the above observation if it is found that activity level is not high, RB coolers and secondary purge system are operating normally, then no immediate manual action is called for.

However, to reduce the iodine activity concentration PCFPB system can be started after some deliberate initial delay. This is to settle down the atmospheric condition in RB and to avoid more iodine loading in filters. Also it may help in large aerosols to settle down, thus prefilter will not be loaded unnecessarily. A delay of about 5.0 hrs is considered in present emergency operating procedure.

Next if continuous compressed air in leakage exists (even after isolating non essential air lines, as mentioned in 2.5) then, depending upon containment pressurization rate operator has to decide whether PCCDS operation is called for. If possible this should be avoided. But in case it is seen that the secondary purge system is not effectively functioning and negative pressure is not developed, in that case to avoid excess ground level release, PCCDS can be operated. Fig.8 (a&b) shows the variation of ground level and stack level releases with respect to delay time of PCCDS operation when there is no secondary purge available.

It is seen that early operation of PCCDS is more effective in reducing ground level release when SCRPS is not operational. However, stack release decreases with increasing delay time. This two opposite trend results in a optimum delay time, where effective dose will be minimum from combined effect of ground and stack level releases.

Also another important factor which operator has to consider is the prevailing weather condition. It is to be noted that normal practice in safety analyses for dose calculation is to consider Pasquill's weather condition -F for ground level release and Pasquill's weather condition -B for stack level release which are most conservative assumptions. But in actual situation if ground level releases stop due to negative pressure in secondary annulus, then operator can delay the operation of the PCCDS, depending on favorable weather condition and the direction of wind flow to minimize the consequence of stack release. By the time operation of PCCDS is called for (typically 24 hours or more), a senior level team of experts will be available to guide the emergency operation, who can take into account the various factors discussed above, including operating status and effectiveness of various ESFs. The operators will be equipped with the software for carrying out the release calculations (based on the prevailing status of ESFs and weather conditions) in computer. Thus the quantitative estimates will help in taking appropriate management strategies.

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ANALYTICAL STUDIES RELATED TO INDIAN PHWR CONTAINMENT SYSTEM PERFORMANCE

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Abstract

Build-up of pressure in a multi-compartment containment after a postulated accident, the growth, transportation and removal of aerosols in the containment are complex processes of vital importance in deciding the source term. The release of hydrogen and its combustion increases the overpressure.

In order to analyse these complex processes and to enable proper estimation of the source term, well tested analytical tools are necessary. This paper gives a detailed account of the analytical tools developed/adapted for PSA level 2 studies.

1.0 INTRODUCTION

The Indian Pressurised Heavy Water Reactor (PHWR) is provided with a suppression pool type of containment designed to withstand the consequences of the loss of coolant accident (LOCA) and the main steam line break accident (MSLBA). In the event of a postulated accident, the steam flashes into the containment, leading to pressure build-up. In case of a severe accident, radioactive aerosols may be released into the containment. There is also a possibility that due to metal-water reaction, hydrogen may be generated and released into the containment.

In order to analyse the performance of the containment system under accident conditions, a programme of analytical and experimental studies is being pursued at the Bhabha Atomic Research Centre. As a part of this programme, the Containment Transient Analysis code CONTRAN has been developed to analyse the pressure and temperature transients in containment, following LOCA and MSLBA. For the analysis of hydrogen mitigation phenomena in the containment using catalytic recombiners, the code HYRECAT has been developed. To analyse the complex processes of transport, growth, deposition and removal of aerosols within the containment the code NAUA/MOD5 has been adapted. For analysing the aerosol scrubbing behaviour of the suppression pool, the code SPARC has been adapted after incorporating suitable modifications.

A detailed account of the development, validation and adaption of the analytical tools for the analysis of the containment response following an accident is presented in the following sections.

2.0 THE CONTAINMENT OF INDIAN PHWRs

The PHWRs of standardised design in India use a double containment (Fig.1), the inner (primary) containment being made of prestressed concrete and the outer (secondary) containment being made of reinforced concrete. The annular gap between the two containments is maintained under partial vacuum to prevent leakage from within to the atmosphere. The primary containment can be further subdivided into two accident based volumes called V1 (drywell) and V2 (wetwell). These two volumes are interconnected by the vent system via the vapour suppression pool. The vapour suppression pool, which passively absorbs energy released during LOCA, also helps in the scrubbing of a part of fission products released, if any, in the event of accidents leading to core damage.

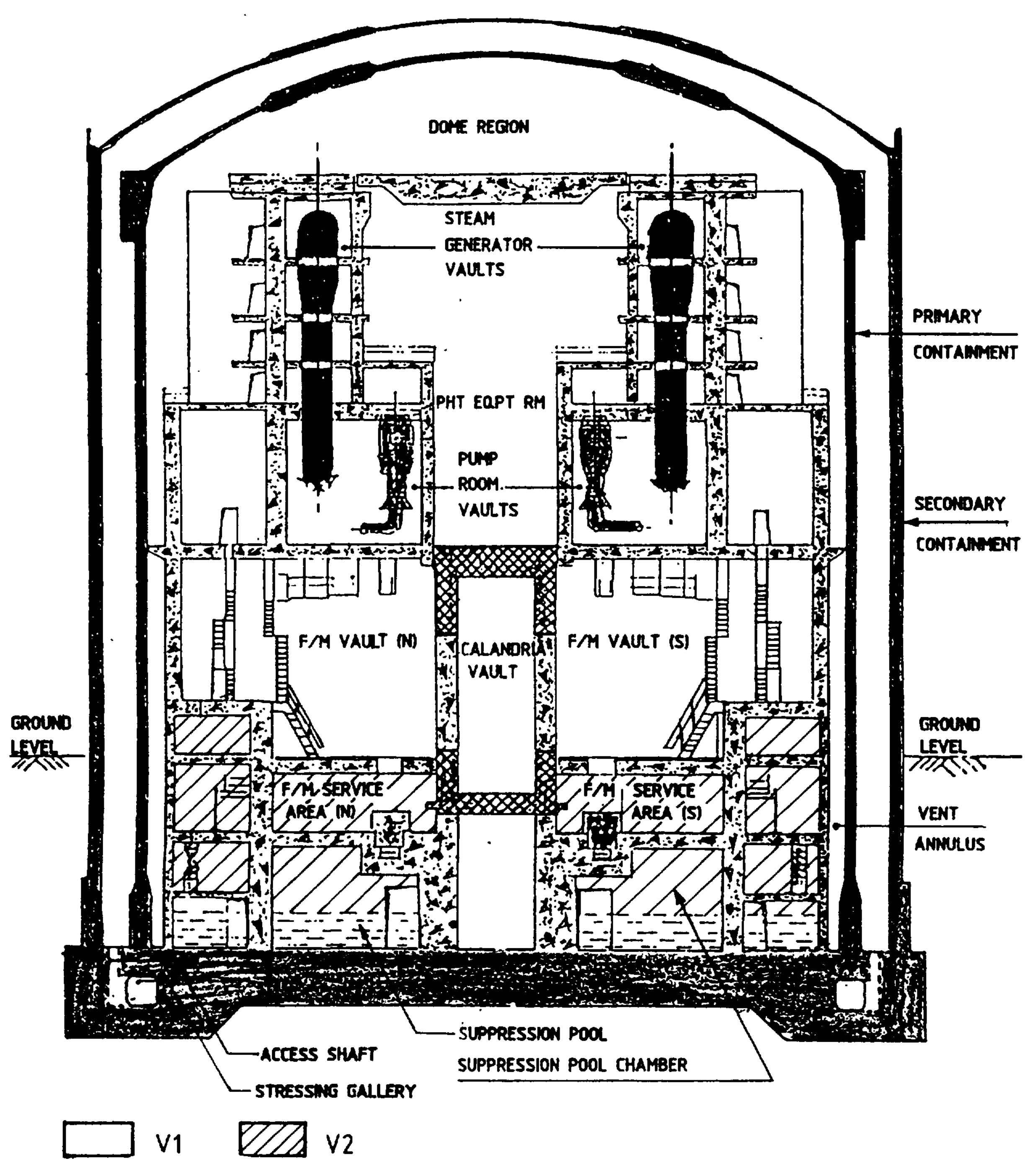


Fig.1: Typical Cross-Section of 220 MWe
Indian PHWR Containment Building

3.0 CONTAINMENT SYSTEM PERFORMANCE STUDIES

The source term for the environment depends among other factors, upon the integrated over-pressure in the containment, the air-borne concentration and characteristics of the aerosols released into the containment and the leak tightness performance of the containment. The containment overpressure could be due to a LOCA or MSLBA. Release of hydrogen and its combustion could also result in further pressurisation. Various natural processes such as heat absorption by the containment structure help in reducing the containment overpressure and aerosol decay and deposition processes help in reducing the air-borne aerosol concentration in the containment, thereby reducing the source term for the environment. In addition, different engineered safety features employed in the plant help in mitigating the consequences of the accident. To analyse these complex processes and to enable proper estimation of the source term, well tested analytical tools are necessary. These tools are either specifically developed or suitably adapted for the specific reactor system and satisfactorily validated. Sensitivity analysis is required to be carried out to quantify the influence of various factors on the containment design parameters. Such validated and well tested computer codes could then be used to analyse and improve the performance of the containment system.

4.0 ANALYTICAL STUDIES ON CONTAINMENT BEHAVIOUR

4.1 CONTAINMENT THERMAL-HYDRAULIC TRANSIENT ANALYSIS

The CONtainment TRansient ANalysis code CONTRAN has been developed to analyse the pressure and temperature transients in the containment, following LOCA and MSLBA. The code has been extensively validated against data from in-house and other experiments. A large number of analyses have been carried out using the code CONTRAN for Indian PHWRs.

4.1.1 Computer code CONTRAN

The multi-compartment containment geometry is modelled in the code by a network of volumes interconnected by junctions. The detailed formulation of the code is described in Ref. [1]. Salient features of the code CONTRAN are as follows:

1. Consideration of various heat sinks and heat transfer processes.
2. Vent clearing transient model for vapour suppression pool based on 1-D momentum equation [2].
3. Differential pressure driven hydrogen transportation model.
4. Hydrogen combustion model based on Adiabatic Isochoric Complete Combustion (AICC) of hydrogen.

4.1.1. Validation of the code CONTRAN

The code has been extensively validated using the test data from various sources as described below.

a. One-Tenth Scale Vapour Suppression Pool Containment Experimental (VSPE) Facility, Kalpakkam, India

The experimental facility (Fig. 2) at Kalpakkam, (India) is a one-tenth scale model of the vapour suppression pool type multi-compartment Indian PHWR containment. A large number of tests with blowdown from a PHT system model were carried out in this test facility to study the influence of various governing parameters on containment behaviour [3,4,5]. One such typical result showing comparison between code predictions and the experimental data in respect of pressure transients in V1 (drywell) and V2 (wetwell) for the test M4LO [1] is shown in Fig.3. The designated test was performed to simulate a large break LOCA in MAPS (Madras Atomic Power Station) reactor.

It can be seen from Fig. 3 that the first pressure peak of 121 kPa, corresponding to pressure buildup in the fueling machine vault till the rupture of blowout panel has been predicted correctly. The subsequent pressure transients and the value of the second pressure peak of 116 kPa (due to pressurisation of V1 until vent clearing) seem to be on the conservative side with respect to the test data. The comparison of CONTRAN predictions with data from other tests are also of similar nature and are generally found to be satisfactory.

b. Battelle Frankfurt Containment (BFC) Test Facility, Germany

The Battelle Frankfurt Containment (BFC) test facility in Germany (Fig. 4), is a one-fourth scale model concrete containment with a total free volume of about 580 cu. m. [6]. The

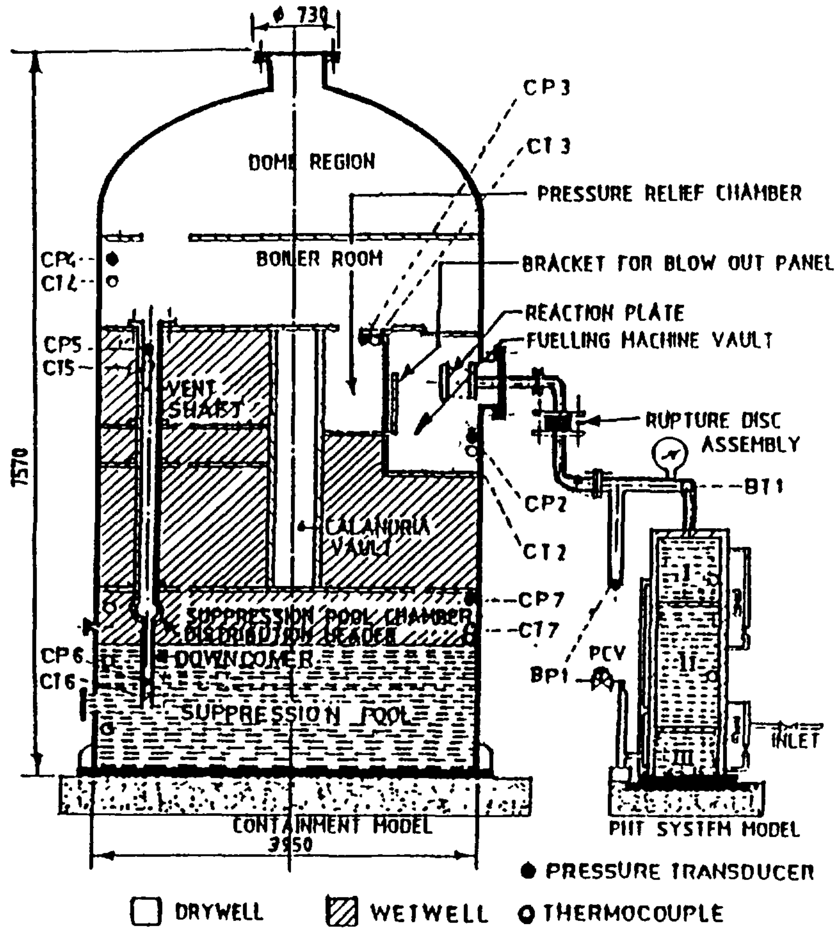


Fig. 2: One-Tenth Scale Vapour Suppression Pool Containment Experimental (VSPE) Facility

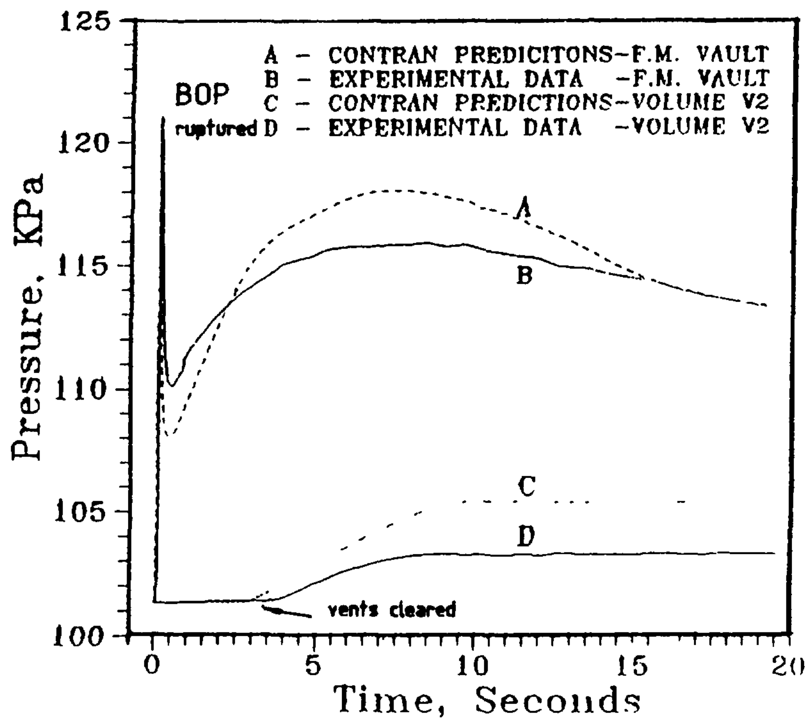


Fig. 3: Pressure Transients in V1 and V2 for Test M4L0 in One-Tenth Scale Vapour Suppression pool Containment Experimental Facility

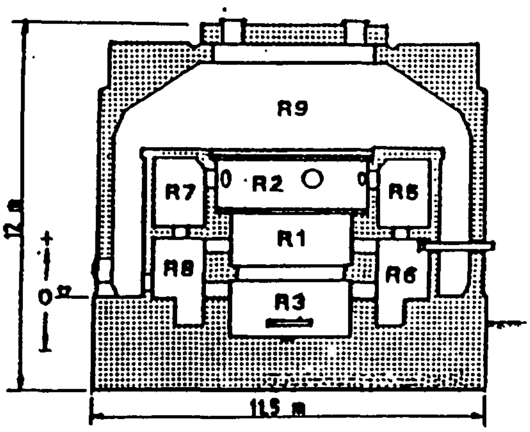


Fig. 4 ONE-FOURTH SCALE BATTELLE FRANKFURT CONTAINMENT MODEL (BFC)

containment model has several compartments with a provision to select different multi-compartment configurations. Validation of the code CONTRAN has been successfully carried out [7,1] using the data from the two blowdown tests designated as BFC Test-D1 and BFC Test-D15 [8]. The code validation results for the Test-D15 are presented in Fig.5.

The Test-D15 involved blowdown of saturated steam for 2.92 seconds from 69.8 bar in room R6 (Break Compartment). The results comparing the CONTRAN predicted pressure transients in the break compartment R6 with the test data up to 2.5 seconds are shown in Fig.5a. The corresponding predictions of COBRA-NC for this test, reported earlier [8], are also shown in Fig.5a for comparison. The pressure transients predicted by the code CONTRAN for the other compartments viz. R8, R7, R4 and R9 are compared in Figs.5b-e, with the experimental data and the COBRA-NC code predictions. The CONTRAN predicted pressure transients are observed to be conservative with respect to the test data.

c. CSNI Numerical benchmark Test

The code CONTRAN has also been successfully validated [9] using the data from the numerical benchmark test [10] devised by the Committee on the Safety of Nuclear Installations (CSNI) of OECD. The benchmark exercise was devised to test the containment analysis computer codes for numerical accuracy and convergence errors in the computation of mass and energy conservation for the fluid and in the computation of heat conduction in structural walls.

The benchmark test model, shown schematically in Fig.6, consists of a single fluid volume into which steam and water are injected and from which heat is transferred to a single concrete wall. Based on the semi-infinite solid heat slab with a specified constant heat transfer coefficient on the inner wall surface, analytical solutions were provided to arrive at the transient wall surface temperature, total pressure and steam partial pressure in the containment. The code CONTRAN successfully predicted all the three transients with good agreement. Fig.7 shows the comparison of predicted transients for total containment pressure (Fig.7a), steam pressure in the containment (Fig.7b) and structural wall surface temperature (Fig.7c) with the test data.

4.1.2 Analysis of Indian PHWR Containment using code CONTRAN

Using the code CONTRAN, a large number of analytical case studies have been carried out to estimate the effect of variation of the following parameters on the containment pressure-temperature transients [11].

i. suppression pool bypass area
ii. surface area of the containment structure
iii. heat transfer to the structural members
iv. multi-compartment configuration
v. initial relative humidity in the containment
vi. initial temperature in the containment
vii. loss coefficient of the vent system
viii. integrated mass/energy discharge into the containment

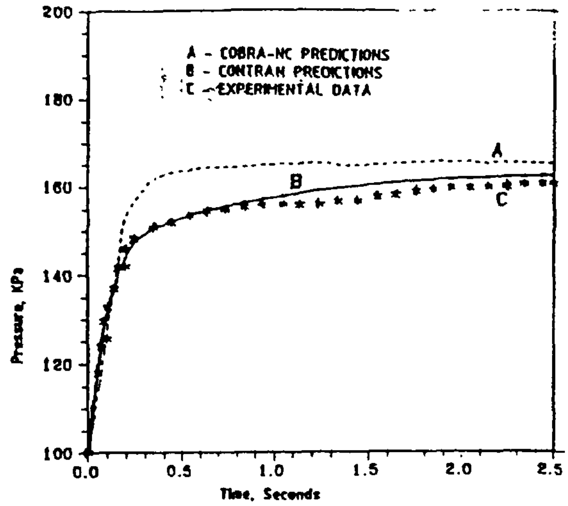


Fig. 5a: Pressure Transients in Room R6 (Break Campartment) for BFC Test D-15

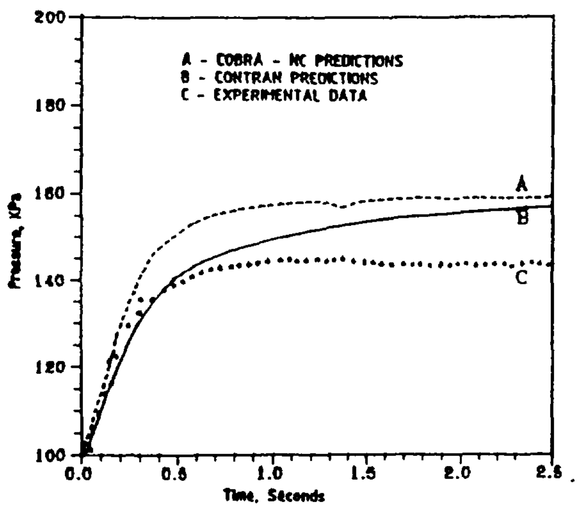


Fig. 5b: Pressure Transients in Room R8 for BFC Test D-15.

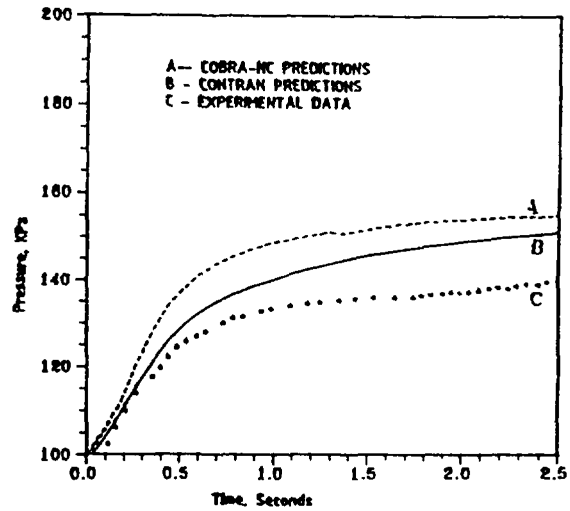


Fig. 5c: Pressure Transients in Room R7 for BFC Test D-15.

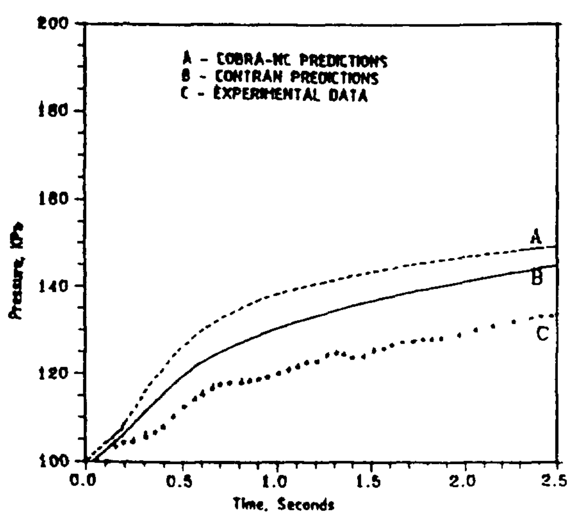


Fig. 5d: Pressure Transients in Room R4 for BFC Test D-15.

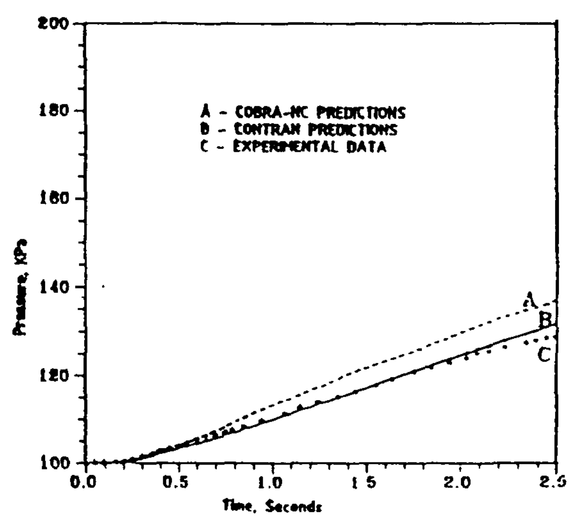


Fig. 5e: Pressure Transients in Room R9 for BFC Test D-15.

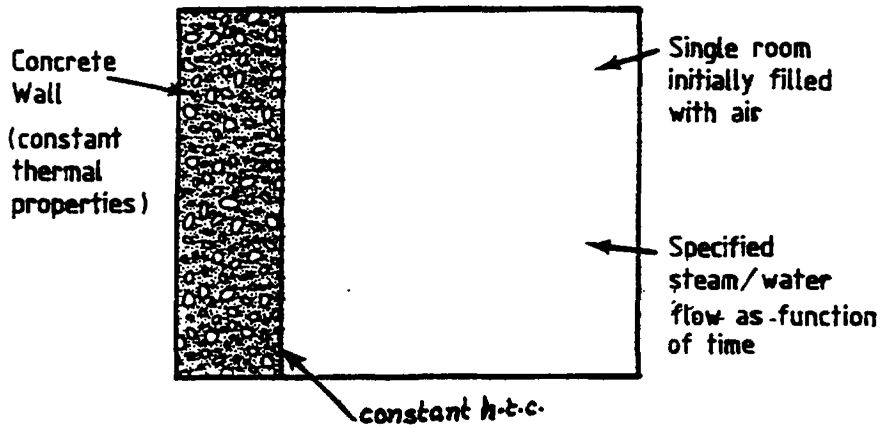


Fig. 6: CSNI Numerical Benchmark Test Model

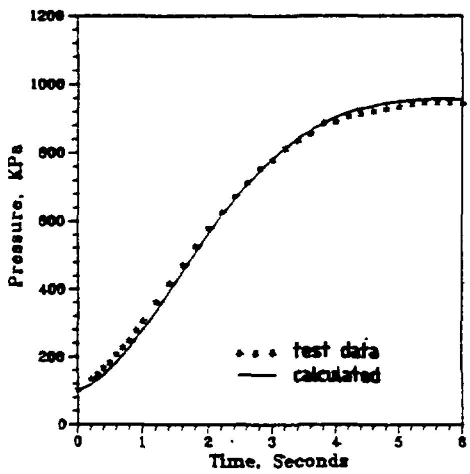


Fig. 7a: Total Pressure in Containment of CSNI Benchmark Test Model

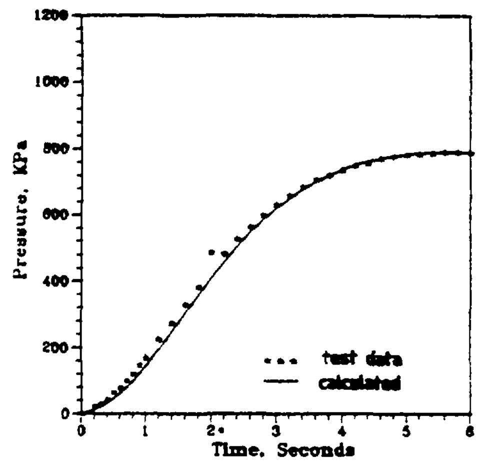


Fig. 7b: Steam Pressure in Containment of CSNI Benchmark Test Model

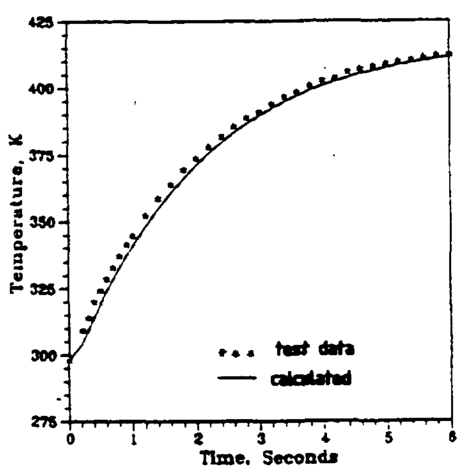


Fig. 7c: Wall Surface Temperature of Containment of CSNI Benchmark Test Model

Results of some of the analyses, e.g. the effect of variation of the suppression pool bypass area (Fig.8a), surface area of the containment structure (Fig.8b), heat transfer to the structure (Fig.8c) and multi-compartment configuration (Fig.8d) on the containment peak pressure are presented.

It has been observed that, the containment peak pressure increases with increase in suppression pool bypass area and decreases with increase in the heat transfer coefficient and surface area of structural heat slabs.

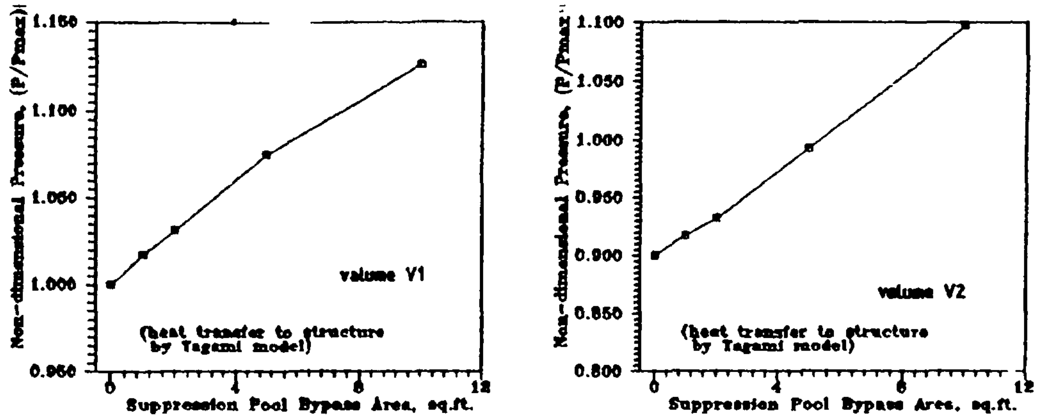


Fig. 8a: Variation of Containment Peak Pressure with Suppression Pool Bypass Area following MSLBA in 220 MWe Indian PHWR

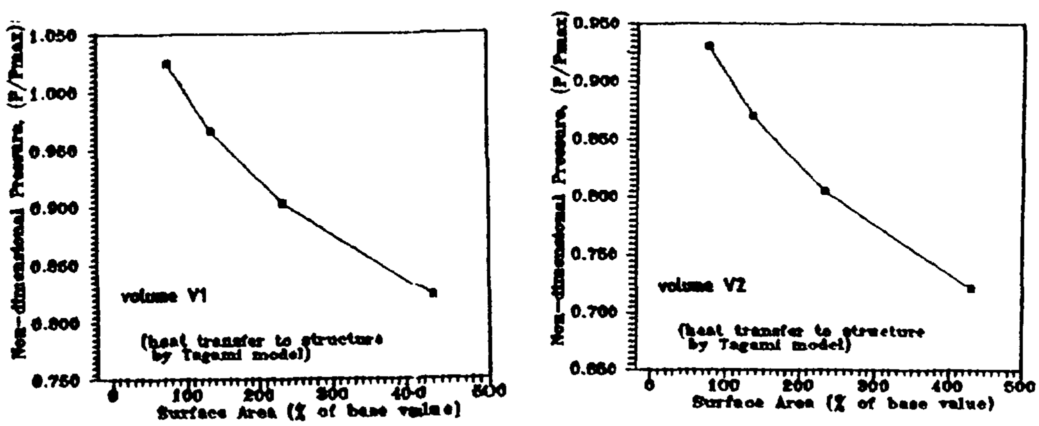


Fig. 8b: Variation of Containment Peak Pressure with Surface area of Structure following MSLBA in 220 MWe Indian PHWR

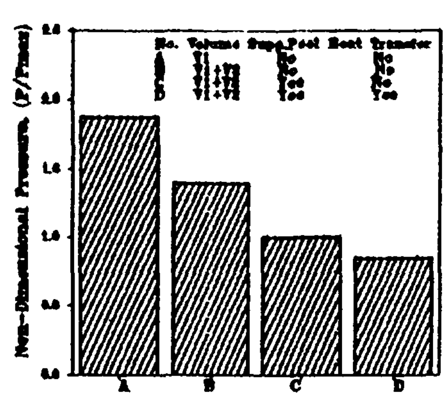


Fig. 8c: Effect of Suppression Pool and Structural Heat Transfer on Containment Peak Pressure following MSLBA

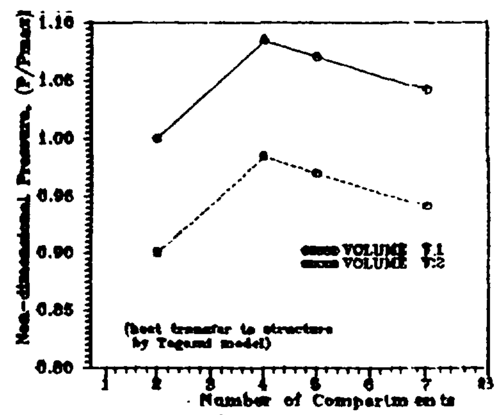


Fig. 8d: Effect of Multi-compartment geometry of Containment on Peak Pressure following MSBLA

The code CONTRAN, is extensively used for studying the performance of the containment of IPHWRs for various LOCA/MSLBA scenarios [11].

4.1.3 Experience gained during validation of the code CONTRAN

Heat and mass transfer processes play an important role in the estimation of the design pressure and temperature of the containment of a Nuclear Power Plant. While using a particular model for the condensation heat transfer processes, the conservativeness of the model should be ensured. The conduction heat transfer in the structure is better modeled by using logarithmic grids, with the grid size increasing from inner to outer surface. The use of uniform grids may result in under-estimation of the containment design pressure. Lower initial relative humidity and lower initial temperature in the containment result in higher design pressure. Sensitivity analysis helps in assessing the influence of various parameters in arriving at a conservative estimate of the containment design pressure.

4.2 HYDROGEN BEHAVIOUR IN CONTAINMENT

Since the accident at Three Mile Island (TMI) Nuclear Power Plant, there has been a great deal of interest regarding the behaviour of hydrogen in the containment. In the event of highly unlikely occurrence of a severe accident in nuclear reactors, a large quantity of hydrogen could be generated. The hydrogen so generated will be confined within the containment. Due to the very large volume of containment of IPHWRs, the global hydrogen concentration is estimated to be lower than the deflagration limit of hydrogen-air mixture. Depending upon the distribution characteristics of hydrogen in a multi-compartment containment system, there could possibly be a formation of local pockets of high concentration. If the deflagration limit is exceeded, it can cause combustion. This exothermic reaction may cause further overpressurisation of the containment. To mitigate the consequences of such a situation, there are currently four approaches being investigated worldwide, viz.

* inertisation of the hydrogen-air mixture,
* deliberate ignition of combustible mixtures,
* catalytic recombination of hydrogen with oxygen [12].
* combination of deliberate ignition and catalytic recombination [13] or simultaneous inertisation and catalytic recombination [14]

Among these, the catalytic recombination method offers promising potentia... its hydrogen removal efficiency even at low hydrogen concentrations and in presence of steam, as has been demonstrated by the reported experimental studies [12].

4.2.1 Computer code HYRECAT

For the analysis of hydrogen mitigation phenomena in the containment using catalytic recombiners, the code HYRECAT has been developed [15,16]. The model is based on the solution of equations for mass and energy conservation within a homogeneously mixed hydrogen-air-steam mixture undergoing oxidation of hydrogen on the catalyst surface. The salient features of the code HYRECAT are as follows.

4.2.2 Salient Features of code HYRECAT

1. The kinetics of catalytic oxidation of hydrogen is described by Arrhenius type empirical rate equation, (on the basis of unit mass of catalyst or unit area of the catalyst coated surface) [17,18].
2. Appropriate heat transfer mechanisms are modelled for the dissipation of the heat of reaction from the catalyst to the surroundings.
3. The code can analyse the deliberate ignition mode of hydrogen recombination and also the dual mitigation system involving catalytic recombiners and igniters together. The transient containment pressure and temperature consequent to the combustion is evaluated by using (a) AICC and (b) laminar burning velocity models [19] alongwith various thermal hydraulic processes.

4.2.3 Validation of code HYRECAT

Bhabha Atomic Research Centre (BARC) has successfully developed a special type of activated platinum catalyst and the laboratory tests have demonstrated its satisfactory performance. Further tests on this catalyst system are planned on an engineering scale test facility (see Fig.9). The validation of the code HYRECAT has been carried out [15] using data from two of the several laboratory tests performed at BARC. The test was performed in a 22 litre stainless steel vessel with a catalyst coated substrate of known dimensions and known catalyst loading (i.e. mass of catalyst per unit area), with initial hydrogen concentration of 5.1 % (v/v). The variation of the total pressure in the vessel was recorded continuously.

Excellent agreement has been observed between the predictions of HYRECAT and the measured total pressure in the vessel as seen from Fig.10. More detailed results of validation are presented elsewhere [15].

4.2.4 Analysis of planned experiments using code HYRECAT

Large scale experiments are planned on a 22 cu.m. vessel to examine the behaviour of the catalytic recombination system on engineering scale (see Fig.9) and to test the performance of the device [20]. The test facility consists of hydrogen injection and monitoring system, catalytic recombiner device, steam and air injection provisions, various instrumentations and the safety devices installed on the model containment vessel. The code HYRECAT was used to analyse these tests to be conducted at BARC and predict (pre-test) the performance of the catalytic recombiner system. The predicted behaviour for a typical test, in terms of the containment pressure, temperature, hydrogen concentration, relative humidity and reaction rate etc. are presented in Figs.11a-f. These tests would generate a data bank useful for further validation of the code.

4.3 AEROSOL SOURCE TERM ANALYSIS

To enable a realistic assessment of the source term, the complex processes of transport, growth, deposition and removal of aerosols within the containment are required to be analysed. The codes NAUA/MOD5 [21] and SPARC [22] have been adapted for this purpose.

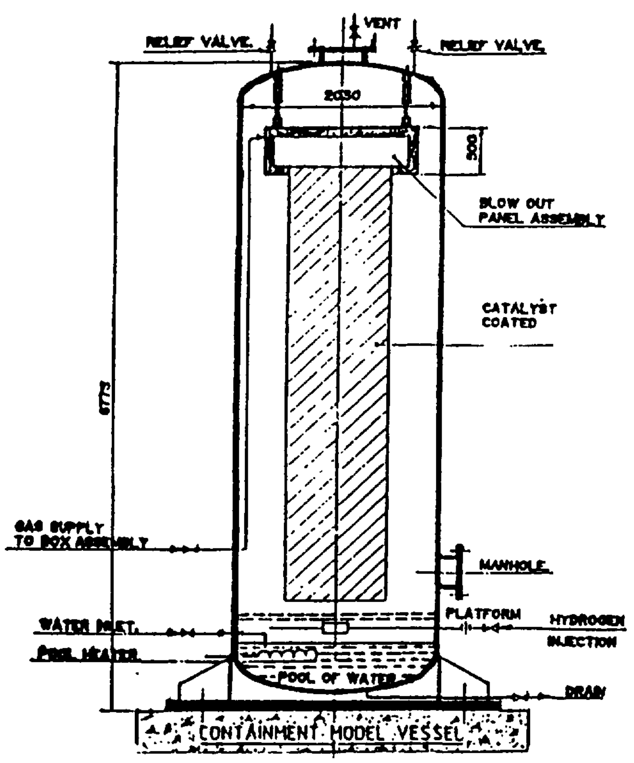


Fig. 9: Experimental Setup for Hydrogen Mitigation Studies using catalytic Recombination

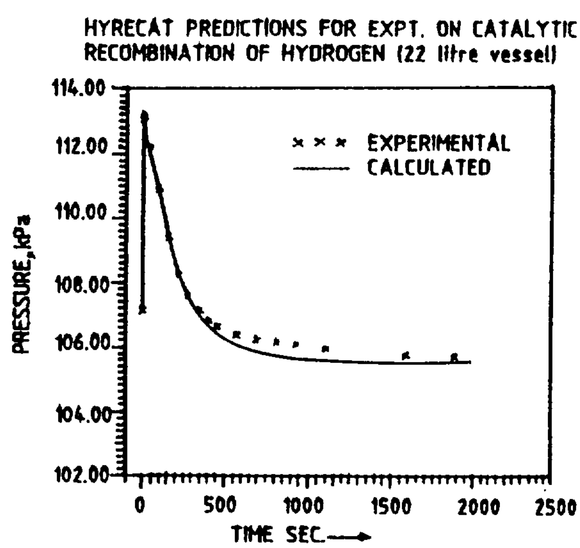
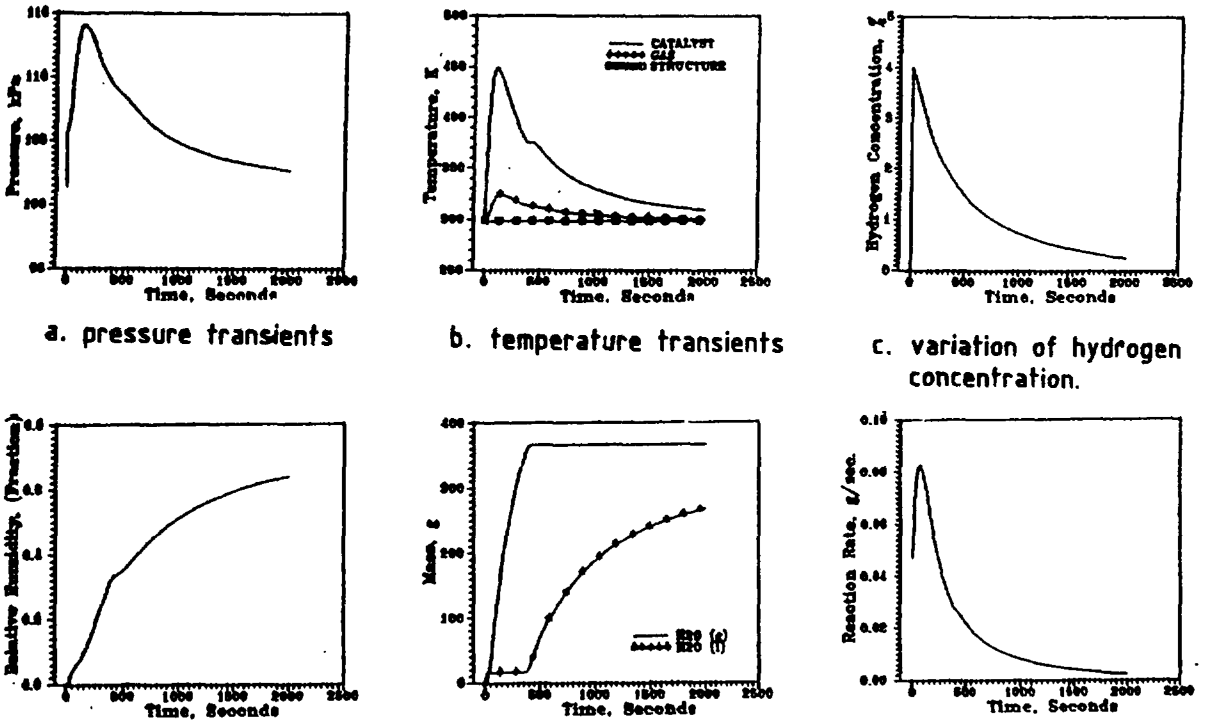


Fig. 10: Comparision of HYRECAT Predictions with experimental data on Laboratory Scale Test.



a. pressure transients
b. temperature transients
c. variation of hydrogen concentration.
d. variation of R. H.
e. variation of H2O mass
f. variation of reaction rate

Fig. 11: Analytical results for the proposed experiment for hydrogen mitigation studies using catalytic recombination on model containment.

4.3.1 Computer code NAUA MOD 5

NAUA is an advanced multi-compartment aerosol behaviour analysis code for use in reactor containments following core-melt accidents. It models the physical aerosol phenomena occurring naturally in the containment. It models gravitational deposition, diffusional plate-out and diffusio-phoresis as removal mechanisms. Among the interaction processes, the code considers Brownian coagulation, gravitational agglomeration and aerosol growth by steam condensation. The code is not integrated with the containment thermal hydraulics. NAUA can take into account the build-up of upto 50 nuclide species on a size-dependent basis.

4.3.2 Assessment of code NAUA MOD 5 with the Falcon Aerosol Test

The Committee on the Safety of Nuclear Installations (CSNI) of OECD organised during 1992-94, the International Standard Problem Exercise no. 34 (ISP-34) on aerosol behaviour in the primary circuit and the containment of a nuclear reactor. The exercise involved carrying out specific experiments in the Falcon Test Facility at Winfrith Technology Centre, U.K. and analysis of these experiments using different computer codes [23].

The tests were performed in the small scale Falcon test facility shown schematically in Fig.12. The facility comprised of core region, primary circuit piping and the containment. Simulant fuel samples clad in zircaloy-4 were placed in a silica vessel simulating the core region. Heating of the fuel was achieved in a 40 kW induction furnace. Helium gas as a carrier medium was admitted at the bottom of the silica vessel and aqueous boric acid solution could be introduced on the heated sample. The thermal gradient tube and the stainless steel pipe simulated the primary circuit. The containment was a 0.3 cu.m stainless steel chamber connected to the primary circuit.

Containment analysis of these tests was carried out at BARC using the computer code NAUA MOD5. The code calculations for one of the experiments, viz. FAL-ISP-1 are presented and compared with the experimental data in Figs.13a-d [24,25]. Many of the code predictions were found to be in reasonably good agreement with the experimental results. The containment floor deposits are predicted closely, whereas the wall deposits are underpredicted. However, the

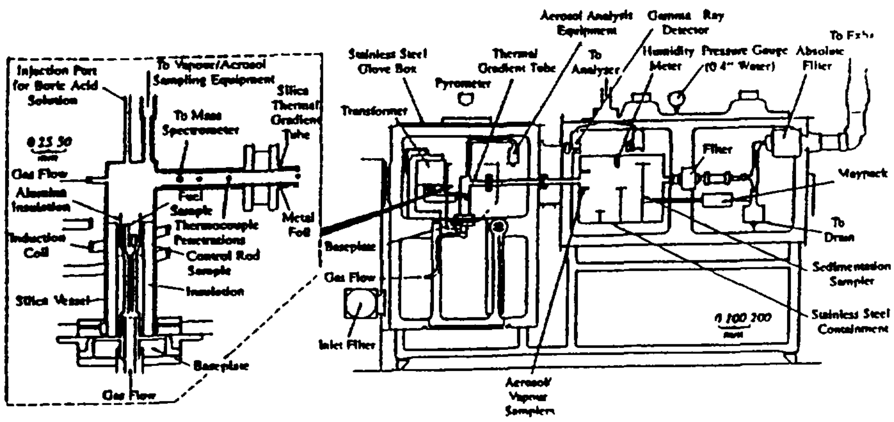


Fig 12· Falcon test facility for Aerosol Studies

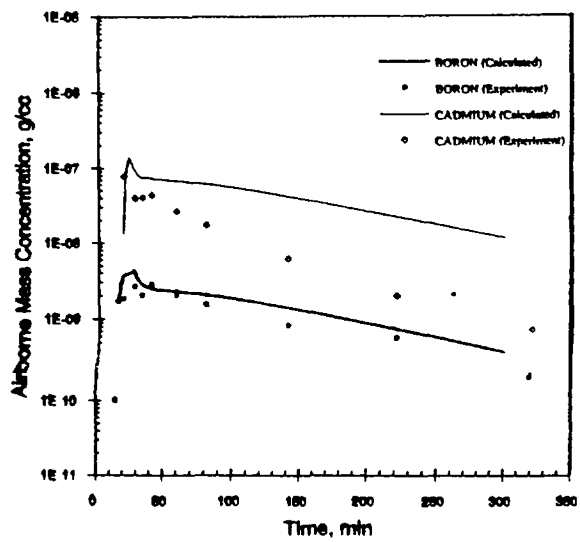


Fig. 13a: Boron and Cadmium airborne mass concentrations in containment

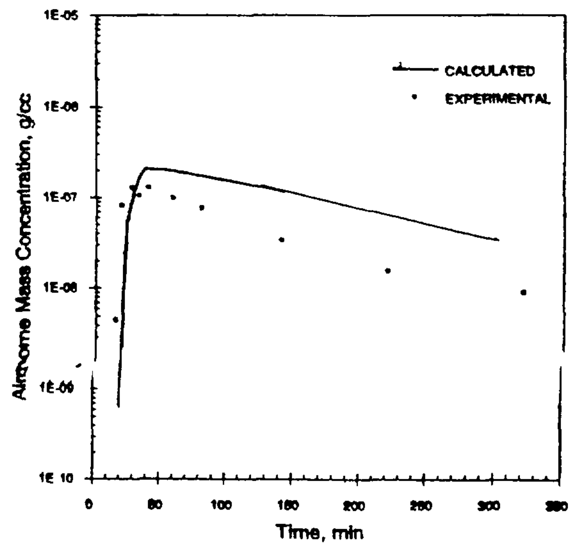


Fig 13b· Cesium airborne mass concentration in containment

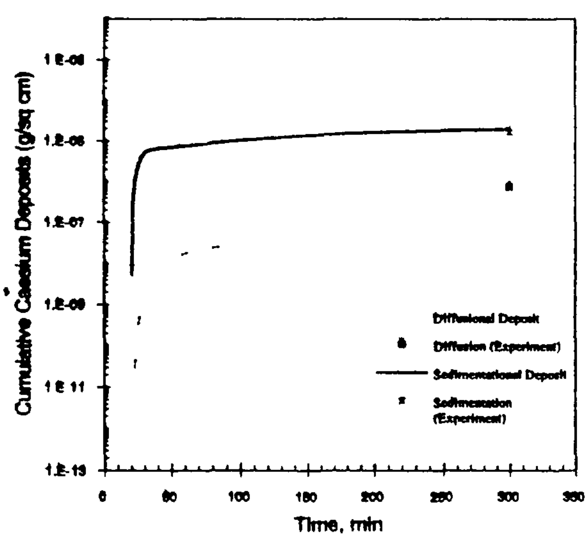


Fig. 13c: Cumulative deposition of Cesium

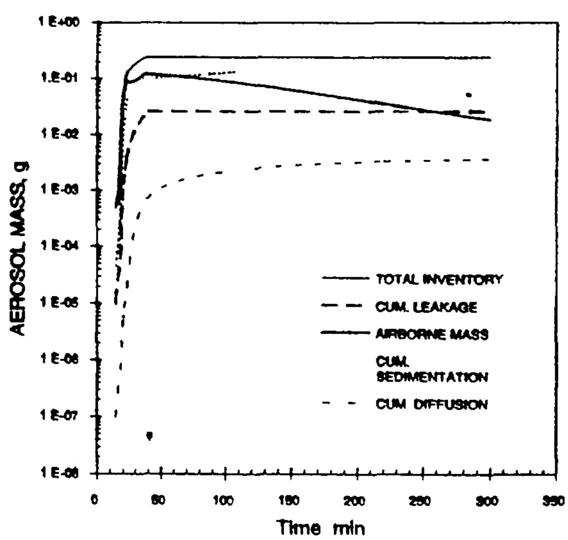


Fig 13d; Removal of aerosols by various mechanisms.

predictions are observed to be well within the reported uncertainties in the experimental measurements. Augmentation of the code for integral thermal hydraulics and assessment of the containment behaviour of Indian PHWRs is in progress.

4.3.3 The computer code SPARC

Engineered Safety Features (ESFs) such as suppression pool are a part of the IPHWR containment system. Besides its intended function of reducing the containment pressure, the suppression pool also helps in the removal of aerosols.

In order to analyse the aerosol scrubbing behaviour of the suppression pool, the code SPARC has been adapted after incorporating suitable modifications. The code models the particle capture by condensation of steam, impaction, sedimentation, centrifugal deposition and diffusional deposition. Growth of soluble particles by water vapour absorption is also modelled. The removal of aerosols is quantified by decontamination factors (DF) defined as the ratio of concentration of aerosols in the gas entering the pool to that leaving the pool [22].

4.3.4 Validation of the code SPARC

Laboratory scale experiments were carried using a prototype bubbler unit (Fig.14). The nebuliser generates wet aerosols from aerosol solution. Compressed air is used for drying these aerosols and carry them to the pool of water. Suction produced by the pump forces the aerosol-laden air to bubble through the pool of water. Aerosols were collected on filters after passing them through the pool and compared with the case without passing through the pool to obtain the decontamination factors. Experiments with two different sized particles viz. Fluorescein (MMAD = 0.8 microns) and Silica (MMAD = 0.2 microns) were carried out. Figs.15a-b show the analytically computed DFs as a function of size classes for Fluorescein and Silica aerosols respectively. The code predictions and experimental results presented in these figures are found to be in close agreement [26].

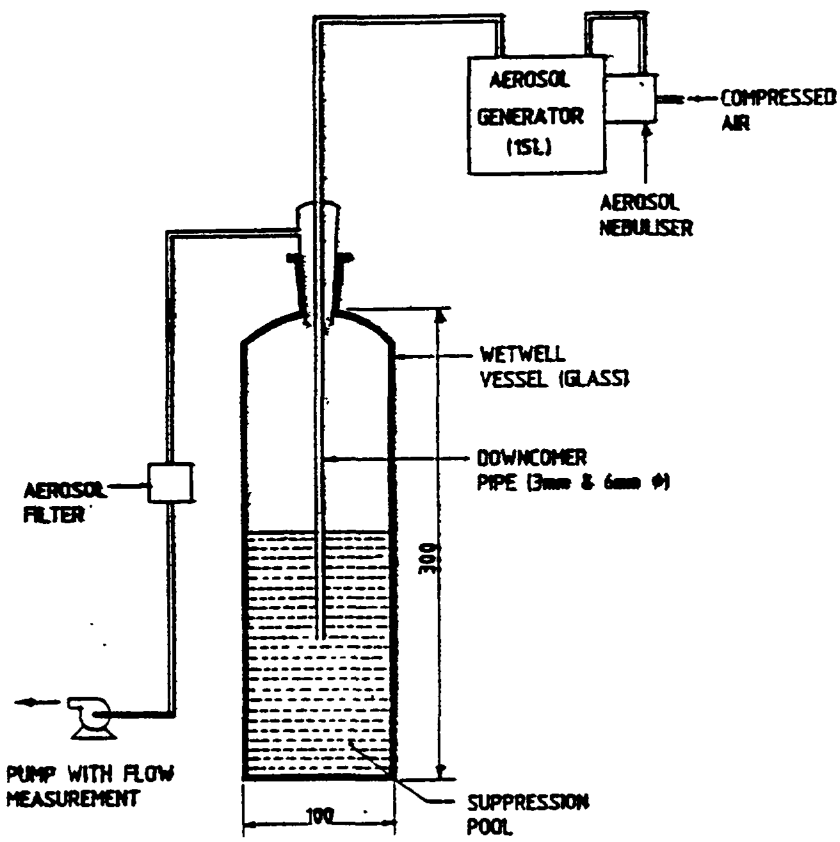


Fig. 14: Suppression pool Aerosol Scrubbing Test Facility (schematic)

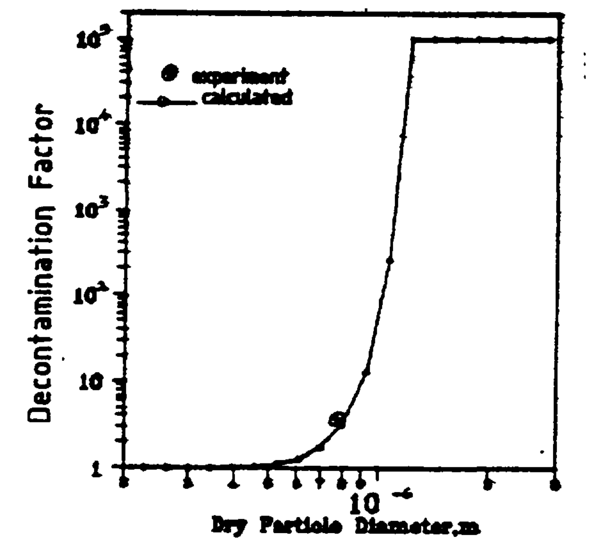


Fig. 15a: Effect of particle diameter
for Fluorescein aerosols

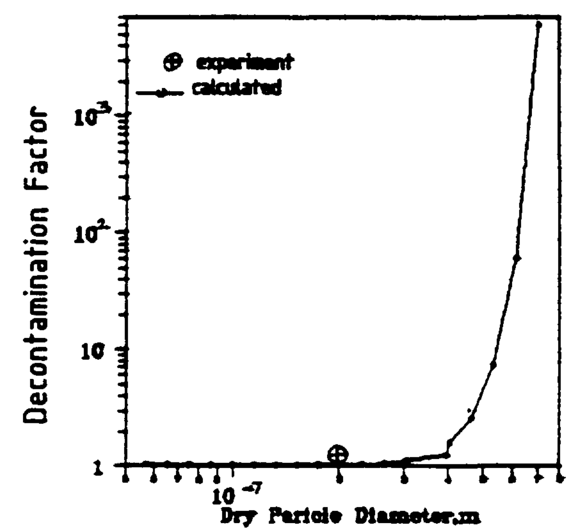


Fig. 15b: Effect of particle diameter
for Silica Aerosols

5.0 CONCLUDING REMARKS

The various codes under use at BARC for analysing containment related scenario following DBA or severe accidents are briefly described. The results of the validation exercises performed to test the code capabilities are also described The code CONTRAN for predicting thermal hydraulic transients in the containment has been validated extensively The codes for hydrogen and aerosol behaviour have been validated to a limited extent. Augmentation of the aerosol codes for integral thermal hydraulics and assessment of the containment behaviour of Indian PHWRs is in progress.

While the various accident scenarios, performance of the protective systems and engineered safety features and the behaviour of the released products decide the gross values of the source-strength and the probability of the associated event, the uncertainties can be estimated from the scatter of the values of various code predictions. The studies aimed at understanding, analysing as well as improving the containment performance being carried out at BARC are presented and discussed.

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APPROACH TO DEVELOPMENT AND USE OF PSA LEVEL 2 ANALYSIS FOR THE CERNAVODA NUCLEAR POWER PLANT

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Abstract

This paper first describes the status of PSA activities for the Cernavoda NPP and the extension of the PSA work to include Level 2 PSA. Important characteristics of this reactor type for Level 2 PSA are outlined. Due to the specific layout of the CANDU reactor the evolution of severe accidents is considerably different to vessel type LWRs. Accidents can be roughly categorized into three categories, "severe accidents" which lead to the loss of core structural integrity, delayed loss of core structural integrity as a consequence of the loss of heat sinks, and fuel channel failures. The current work for modelling accident progression in the core region is described. The elements for the Level 2 PSA including definition of PDSs, probabilistic containment logic and source term calculation are outlined. It is pointed out that uncertainties have to be considered which are contained in the models to bridge knowledge gaps. For this purpose sensitivity studies will be carried out for key modelling assumptions.

1. INTRODUCTION

The commissioning phase of Cernavoda Unit 1 has been completed and the connection to the grid for commercial operation has been performed at the beginning of December 1996. During this year it was also decided to restart the unit 2 construction with a tentative schedule for completion in 2000.

PSA activity for Cernavoda Nuclear Power Plant was initiated in 1988 based on the recommendation and significant assistance of IAEA. Now, a full scope level 1 is completed and reviewed by an IAEA IPERS mission. The implementation of review mission recommendations as well as the updating of the study to the as-built documentation for Unit 1 are scheduled for June 1997. During the past 2 years some activities related to PSA level 2 analysis have been initiated.

Since 1990, the safety authority (CNCAN) adopted a specific approach on using PSA methods for the Cernavoda NPP Unit 1 as a licensing documentation. The approach had the goal to enhance the role of the probabilistic methods in the safety documentation in a more detailed manner in order to be more effectively used during the Cernavoda NPP Unit 1 licensing process.

A PSA level 1 study is at present a prerequisite for Cernavoda NPP Unit 1 license renewal after the initial 18 months of operation. In the same time, this study is used as a support documentation in the review and assessment activities for Reliability Analysis (RA) and Safety Design Matrices (SDM) to be issued for licensing the Cernavoda NPP Unit 1.

Detailed licensing requirements for the Cernavoda unit 2 are now under consideration and these include the PSA level 1 and 2 analysis.

2. STATUS OF PSA ACTIVITIES

In 1988 a limited scope Level 1 PSA (Cernavoda Probabilistic Safety Evaluation, CPSE - Phase A) was started and carried out by the Institute for Nuclear Research (INR) in Pitesti with the participation of the designer group represented by CITON. The study, contains 9 event trees and fault trees for 17 systems.

Phase A was completed in 1990 and reviewed by an IAEA International Peer Review Service (IPERS) mission, undertaken within the ROM/91003 project as part of the IAEA Regular Program of Technical Cooperation. The IPERS review has identified a number of issues to be considered in the next phase of the CPSE Study and provided valuable conclusions and detailed recommendations to be followed and implemented.

The CPSE Study restarted in 1993 as a full scope Level 1 PSA, known as CPSE - Phase B. Coordinated by RENEL, Safety and Licensing Compliance and performed by an enlarged PSA team, including CITON Bucharest - Magurele and INR - Pitesti, this new phase of the study benefitted by IAEA assistance under ROM/9/008 Technical Cooperation Project.

Completed in 1995 this phase of the CPSE has been reviewed by an IAEA IPERS mission and the implementation of the review missions recommendations as well as the updating of the study to the as-built documentation for Unit 1 are scheduled for June 1997.

Activities related to PSA level 2 analysis have been initiated in 1994 at the Institute of Nuclear Research in Pitesti and CITON Bucharest (Renel Center for Nuclear Engineering and Technology).

The first step was the familiarization with the Source Term Code Package (STCP), provided by the IAEA, the implementation of the code on PC computers and running some reference test cases.

Furthermore, development of preliminary models, specific for CANDU, and their implementation in MARCH3 has been performed. These models includes the residual heat transfer and cooling degradation of fuel channels, fuel channel debris thermal behavior, moderator thermal hydraulics, and the effect of melted debris on calandria vessel wall.

Based on these models we performed some calculations of different accident scenarios. In reference /3/ the results of a loss of service water initiated accident are presented. The initiated event has as consequences loss of the shutdown cooling system (SDCS), loss of cooling to the moderator, loss of cooling to the calandria vault water, and loss of ECCS heat exchangers. Simultaneously loss of service water, loss of class IV power and loss of steam generator heat sink are assumed.

During 1996, the fault trees for containment systems (containment isolation system, dousing system, reactor building air coolers and ventilation system) have been developed.

Beginning with the next year, a systematic research activity related to hydrogen behavior in containment is initiated. For 1997 a state of the art investigation is planned (hydrogen generation, transport and combustion) as well as a preliminary model development for hydrogen generation.

PSA level 2 activity is benefiting by a significant IAEA support. The fellowships carried out at Korea Atomic Energy Research Institute provided a good level of competence necessary for the initiation of this project. These were particularly useful for providing the technical background necessary for performing the main tasks of the project and the familiarization with the computer code MAAP4 used for CANDU, as well as a specific training related to plant models and methods for physical processes, definition of plant damage states, containment event trees development and source term analysis familiarization.

For the next 3 years it is planned to perform a complete level 2 analysis for the Cernavoda NPP in order to be used in licensing process of the plant and to provide a technical basis for the evaluation and improvements of containment performance.

3. CANDU 600 SPECIFIC FEATURES

The Candu reactors are heavy water-moderated, natural uranium-fueled, and pressurized heavy water-cooled.

Reactor and Primary Heat System

The Calandria is a horizontal cylinder. It is spanned by the fuel channels and contains the moderator. Each fuel channel consists of a pressure tube surrounded by a calandria tube with CO_2 gap in between. Heavy water is used as the coolant and flows at high pressure through the pressure tubes. The moderator is isolated from the hot pressure tube by means of the calandria tube and as a result can be kept relatively cool and can act as a heat sink under certain accident conditions. The fuel element contains natural UO_2 pellets in a Zircaloy-4 sheath.

In case of simultaneous pressure tube/calandria tube rupture, the calandria has been provided with rupture discs to adequate discharge the heavy water flow into the containment.

At each end of the calandria there is an end shield, which contains carbon steel balls and demineralized water in order to provide biological shield and cooling. The calandria assembly is supported by the calandria vault which is filled with light water. This water can act as a passive heat sink. In case of an accident that leads to the corium collection in the calandria bottom, the vault water provides the ex-calandria cooling. The end shields and calandria vault cooling is provided by means of a system. If moderator cooling fails then this cooling system can delay melt relocation into the calandria vault.

The primary heat transport system (PHTS) consists of two closed loops and as a consequence in the event of a loss of coolant accident the rate of the reactor coolant blowdown is reduced. After LOCA the loops are isolated and the intact loop inventory can be maintained.

Special Safety Systems

Emergency Core Cooling System (ECCS)

The basic function of the ECCS is to provide an alternate means of core cooling in case of an accident which depletes the normal coolant inventory in the primary heat transport system to an extent that fuel cooling is not assured. After LOCA, on PHTS low pressure signal (so-called initiating signal) the loops are isolated and if one of the three conditioning signals occurs: high containment pressure or high moderator level or sustained PHTS low pressure, ECCS is automatically initiated by the LOCA signal. The same LOCA signal opens with a delay the main steam safety valves to provide the steam generators crash cooldown.

ECCS consists of three stages according to the operating pressure: High, Medium and Low Pressure Injection.

Shutdown System (SDS)

CANDU reactors are provided with two independent shutdown systems. SDS#1 provides shutdown of the reactor by releasing 28 gravity shutoff rods into the core. SDS#2 can do the same thing by the rapid injection of a gadolinium solution into the bulk moderator. Both systems are designed to have no effect in tripping the reactor in case of failure of any support system.

If failure of both SDS#1 and SDS#2 is postulated when they are needed during a transient, the sequence would be very severe due to the CANDU characteristic of positive coolant and moderator temperature coefficient. However, the failure probability of SDS is very low due to the CANDU design features.

Containment Dousing System

The dousing system function is to limit the magnitude and duration of containment over-pressure caused by a LOCA or a MSLB (main steam large break) inside the containment. The system consists of the dousing tank (DT) located below the dome of the reactor building and six independent spray units connected to the DT.

Reactor Building Air Coolers

One of the main functions of the reactor building cooling system is to provide a long term heat sink (dousing system is the short term heat sink) following a LOCA or a MSLB reducing containment pressure and temperature.

Containment Envelope

The containment envelope is designed to restrict the release of radioactivity to the environment within the maximum permissible dose limits in the event of a radioactivity release. The containment envelope consists of: the reactor building; the airlocks; the containment isolation system.

Feedwater and Steam Systems

Each primary loop is provided with two steam generators. Each steam line is equipped with an atmospheric steam discharge valve (ASDV) in order to release steam when the turbine condenser is unavailable. Sixteen main safety valves (MSSVs) are provided to protect steam generators from over-pressure. The condenser steam discharge valves (CSDVS) discharge the main steam directly to the condenser avoiding the actuation of the MSSVs under severe transients.

Feedwater system supplies water from the deaerator to the boiler level control system via the main feedwater pump lines or the auxiliary pump line in order to maintain the four boiler at required level during various modes of operation.

4. PLANT MODELS AND METHODS FOR PHYSICAL PROCESSES

The first step in Level 2 PSA is to perform plant familiarization and to determine those plant features most likely to impact the progression of the severe accidents. It is necessary to review plant design information, existing CANDU PSA data and severe accident data.

The following most significant areas of the plant will be reviewed:

- engineered safety features which include shutdown cooling system, emergency core cooling system, dousing system, reactor building air coolers;

- interior design of the containment: interconnecting pathways between different containment compartments and the location of equipment will be reviewed to determine major heat sinks, to assess circulation flows within containment and to aid the assessment of hydrogen burning;

- containment pressure boundaries will be reviewed to assess containment failure locations and modes.

- calandria and calandria vault design including pressure relief path from calandria and/or calandria vault and moderator/shield cooling system design.

- service building design will be reviewed to assess the potential for radionuclide attenuation following an interfacing LOCA or a containment leak/break into the service building if the attenuation is possible.

- emergency operating procedures.

All those features which are identified to be able to circumvent potential severe accident uncertainty should be clearly described.

CANDU operation data, severe accident experimental data for CANDU and results from specific computer codes analyses will be reviewed to get insight.

As Candu has its own specific structure the terminology used in the Level 2 PSA study is different from that for PWR:

- **Severe Accident:** loss of core structural integrity, and severe core melt occurrence. Example: LOCA followed by the ECCS and moderator unavailability.

- **Loss of Core Structural Integrity:** loss of heat sinks leading to core melt involving fuel channels failures.

- **Fuel Channel Failure:** simultaneous failure of the pressure tube and calandria tube.

- **Containment Envelope:** comprises the reactor building, sealed penetrations, closed and open penetrations.

To analyze the accident progression means to track the progression of the accident from the onset of core melt until it is assured that no additional release of radionuclides from the containment will occur. In order to achieve this task a plant model is necessary to be developed to simulate the response of the CANDU plant from accident initiation to the source term models. The code MAAP-WS developed for the Wolsong NPP includes models for thermal hydraulics, fission products behavior in the PHTS, calandria, calandria vault and containment during severe accidents.

Another important task is to prepare the plant data which is necessary to run the code. The data can be categorized into generic data which is invariant among the sequences and sequence dependent data which are in concordance with the scenario. Generic data includes plant geometrical data, initial conditions, operational setpoints, engineering safeguard characteristics, fission products initial inventories, etc. Sequence dependent data includes sequence specific information such as: break size and location, what and when operator action are taken, etc. To prepare these data, the accident sequence should be identified in plant damage- event trees and containment event trees in advance.

If the above tasks are done, analysis of the physical progression of an accident using deterministic models is performed to support the following activities:

- Level 1 success criteria evaluation
- event timing determination
- sensitivity studies to support CET development and quantification
- input preparation to source term analysis.

5. PLANT DAMAGE STATES

The interface between the Level 1 system analysis and the Level 2 containment analysis is the classification of each accident sequence into plant damage states (PDS). The purpose of this classification is to reduce the number of the accident analyses required while retaining the essential spectrum of probable accident progression. To group the large number of core-melt sequences into a set of states or bins it is necessary to define a set of functional characteristics of system operation that are important to the accident progression, containment failure and source term analyses.

Level 1 event trees will be extended to include the status of the containment safeguards. The quantified results of extended Level 1 event trees will be propagated through frequencies of the PDSs.

Plant Damage State Grouping Parameters

PDSs are the entry points of the CETs. The following parameters are important for the PDS characteristics: accident progression in the containment; time, mode and location of the containment failure and the radionuclides source terms. The following parameters are proposed to be selected:

- *Containment Bypass* which divides the Level 1 sequences into by-pass and non-bypass groups.
- *Containment Isolation Status* which segregates the Level 1 sequences into distinct groups based on the status of the containment building isolation at the time of core melt.
- *Transient or LOCA* which makes distinction between transient and LOCA sequences.
- *Dousing Spray* which determines whether the dousing spray operates.
- *Shield Cooling* which determines whether this system operates.
- *Local Air Coolers* which determines whether this system operates.

Plant Damage State Definition

Plant damage states are defined by combinations of the possible values for each of the PDS parameters. These combinations are reviewed to delete combinations which are considered not to be physically possible or are counter to other definitions used in the analysis.

The PDS logic diagram is a useful tool for combining the various grouping parameters into unique PDSS. This diagram is constructed with PDS grouping parameters as decision branches to aid in the assembly of specific plant damage state characteristics from the matrix of all possible combinations allowed by the PDS grouping parameters.

6. CONTAINMENT EVENT TREES

Containment event trees (CET) are used to model the containment response by depicting the various phenomenological processes, containment conditions and containment failure modes that could occur during severe accidents.

The integrity of the containment is one of the most important issues in evaluating radioactive releases. Knowledge of performance of the containment structure to internal pressure and temperature transients associated with the accident is essential in determining probabilities of failure and estimating consequences of such failure. The PSA approach determines the median internal pressure capacity at which failure of the containment will occur and the associated variability from which the conditional probability of failure for a given level of internal pressure can be estimated. Failure of the containment is defined as incipient leakage and release of radioactivity material into the atmosphere.

CANDU Containment Design Features

The reactor building is designed in accordance with AECL Design Guides. The containment structure is a prestressed concrete building consisting of three structural components: a base slab, a cylindrical wall and a spherical segmental dome.

To assist in leakage control and cleaning, the containment structure has an internal lining comprising of a plastic coating applied to the inner surfaces of the concrete outer dome and walls.

Beneath the outer dome there is a second dome having an opening in the crown, which together with the perimeter wall forms a container to provide storage of water for dousing and emergency core cooling.

Systems closed within containment, which pass through the containment boundary, and which are open to the atmosphere outside containment, are part of the containment envelope. Similarly a system closed outside containment, and open to the atmosphere inside containment, is part of the containment envelope.

The containment structure is designed to withstand an internal pressure of 0.125MPa.

Method of containment ultimate capacity evaluation

The Probabilistic safety assessment approach for the containment overpressure evaluation is based on a simplified fragility model. The tasks of a containment internal pressure fragility evaluation consist of:

- Identification of potential failure modes

- Development of containment fragilities

- Estimation of leak areas.

Containment event tree

CETs provide a quantitative logic model for examining the spectrum of severe accident progressions and they provide the framework for evaluating the deterministic outcomes of specific accident progression.

The event tree technique is used in modelling containment response. The CET should be constructed in sufficient detail to address the occurrence of various phenomenological events, processes, and human interaction that have been determined to be important to containment failure and source term evaluation. In order to be more scrutable, a CET with a reduced number of events should prove adequate. Complex phenomena can be treated in detail in the Decomposition Event Tree (DET) (or in the phenomenological fault trees).

Each PDS end point represents a unique accident progression starting point with respect to the CET.

It follows that a CET must be developed and quantified for each PDS. In practice, there will be many commonalties for most accident sequences, except for the containment bypass sequence and containment isolation failure.

The events in the CET should contain only those phenomena expected to have the greatest impact on the containment accident progression. Events that contribute to these events and/or which aid in the assessment of the event branch probabilities should be relegated to DETs (or fault trees). Each top event has its own DET (or fault tree). DET consist of the important subevents that contribute to the top event and is used to quantify the branch probability of the top event.

7. SOURCE TERM ANALYSIS

A source term is the amount of radioactivity material which is released to the environment, along with information on the timing, energy, duration, and location of release point.

The CETs produce a large number of end states, some of which are either identical or similar, in terms of key release attributes. The analyses of all accident sequences of interest are prohibitively expensive and time-consuming. As a result the sequences are categorized by their radiological characteristics and potential off-site consequences in what we usually call 'release categories/bins'.

A set of release categories is then defined such that all accidents assigned to the same category are assumed to have the same release fractions of fission products. These release groups are defined on the bases of appropriate attributes that impact fission product releases. Attention should be paid to the requirements of dose calculation as the interface between analysis and dose calculation. To set up the release categories, binning parameters are selected. These parameters will be used as the headings of the source term category diagram. In order to finalize the list of the parameters, previous CANDU and PWR PSA reports will be reviewed and additional plant-specific studies must be performed.

The radiological source terms divide the radioactive material into several source term groups whose elements have similar chemical, physical and physiological properties. In the PWR study the 60 isotopes are placed in nine radionuclides classes which are treated individually in the source term analysis: Inert Gases, Iodine, Cesium, Tellurium, Strontium, Ruthenium, Lanthanum, Cerium, Barium. In case of CANDU reactors, tritium must be considered differently from PWR, as it is produced outside the fuel and is present in the primary heat transport coolant and in the moderator coolant during normal operation.

The objective of the source term analysis is to provide a radiological source term for each release category. In order to estimate the source term, the following steps are usually performed:

1. the CET end-points grouping, which would lead to similar potential radionuclide releases into release categories;

2. determination of the in-core fission products inventory at the initiating time of the accident;

3. determination of the physical progression of the representative accident sequences for each release category;

4. determination of the rate of radionuclides released from the fuel, their transport and attenuation behavior in plant systems and ultimate rate of release from the plant.

In Level 2 PSA studies, considerable uncertainties in source terms exist and care should be taken in using the estimated source terms. A major source of uncertainty are code models due to lack of knowledge. To reduce these uncertainties, sensitivity studies changing the model, assumptions and parameters should be made.

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PSA LEVEL 2 ACTIVITIES FOR RBMK REACTORS

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Abstract

Probabilistic safety analyses (PSAs) of the boiling water graphite moderated pressure tube reactors (RBMKs) have been developed only recently and they are limited to Level 1. Activities at the IAEA were first motivated because of the difficulties to characterize core damage for RBMK reactors. Core damage probability is used in documents of the IAEA as a convenient single valued measure, for example for probabilistic safety criteria. The limited number of PSAs that have been completed for the RBMK reactors have shown that several special features of these channel type reactors necessitate revisiting of the characterization of core damage for these reactors. Furthermore, it has become increasingly evident that detailed deterministic analysis of DBAs and beyond design basis accidents reveal considerable insights into RBMK response to various accident conditions. These analyses can also help in better characterizing the outstanding phenomenological uncertainties, improved EOPs and AM strategies, including potential risk-beneficial accident mitigative backfits. The deterministic efforts should be focused first on elucidating accident progression processes and phenomena, and second on finding, qualifying and implementing procedures to minimize the risk of severe accident states. The IAEA PSA procedures were mainly developed in view of vessel type LWRs, and would therefore require extensions to make them directly applicable to channel type reactors.

1. BACKGROUND AND SUMMARY

Probabilistic safety analyses (PSAs) of the boiling water graphite moderated pressure tube reactors (RBMKs) have been developed only recently and they are limited to Level 1. The limited number of PSAs that have been completed for the RBMK reactors have shown that several special features of these channel type reactors necessitate revisiting of the characterization of core damage for these reactors. The existing characterization of core damage as described in the available guidelines is largely formulated in view of vessel type light water reactors.

In RBMKs, due to their design, and depending on accident conditions, fuel damage could remain either localized, or propagate to become reactor-wide. In addition, it appears, that limited localized damage can be accommodated by design, and would not constitute a large scale release of radioactivity to the reactor coolant system and the confinement building. There would be only small environmental consequences in such cases.

The knowledge base for deterministic assessment of severe accidents in RBMKs is fairly limited and large phenomenological uncertainties remain. It is important to elucidate and to improve the basis for the PSA in areas such as:

- Initiation, progression and propagation of fuel channel failure;

- Grace time and possible emergency operating procedures (EOPs), for example for recovery of a loss of onsite electrical power, including the use of various accident management (AM) strategies to prevent or limit severe accident consequences (e.g.

use of alternative, non-safety grade water sources, reactor coolant system depressurization, etc.);

- Determination of the capability of the ALS (Accident Localization System) system to accommodate severe accident thermohydraulic and radiological loads, taking into account bypass paths through leakage from confinement compartments to the atmosphere. Significant backfitting o[f] existing confinements are planned. This backfitting should include consideration of PSA level 1 results and analysis of severe accidents.

The deterministic efforts should be focused first on elucidating accident progression processes and phenomena, and second on finding, qualifying and implementing procedures to minimize the risk of severe accident states. The IAEA PSA procedures were mainly developed in view of vessel type LWRs, and would therefore require extensions to make them directly applicable to channel type reactors.

2. IAEA CONSULTANTS MEETING ON CORE DAMAGE STATES FOR NPPs WITH RBMK REACTORS

2.1. Motivation and purpose

Probabilistic Safety Criteria (PSC) are often defined in terms of core damage. Examples can be found in the INSAG (International Nuclear Safety Advisory Group) document INSAG-3 [1] and in the IAEA's document on "NPPs built to earlier standards" which is currently under preparation. Using core damage probability as risk measure for a nuclear facility has the following advantages:

- For vessel type LWRs core damage probability is widely used as risk measure and for comparison with other plants;

- For vessel type LWRs core damage probability is a single valued measure which is easy to use and well suited for display of results;

- The scope of many PSAs is limited to Level 1 (analysis of accident sequences up to core damage).

Core damage probability as a risk measure does however not account for release, transport and retention mechanisms of the radioactive materials in the plant. This represents a limitation because for many of the NPPs a large part of the severe accidents are not expected to cause major offsite releases of radioactivity due to the mitigation capacities of plant systems and of the containment. For channel type reactors core damage is more difficult to define as a one valued measure. An issue discussed was therefore whether it is possible and necessary to develop a single valued Level 1 PSA measure for channel reactors or whether it is sufficient or more appropriate to formulate PSCs and make comparisons on a higher PSA level, such as Level 2 (source terms) or Level 3 (consequences).

The primary purpose of the meeting was to discuss the meaning of the term "core damage" for the channel type reactor of RBMK NPPs and to examine existing proposals for the categorization of core damage states. The current definition of core damage in documents of the Agency largely relies on the severe accident phenomenology of vessel type light water reactors with compact cores and no major variants and levels of core damage. In contrast, for the RBMK cores with their large number of relatively detached individual technological channels, there is a possibility of fuel damage in only one or a very small number of channels while the other channels could remain unaffected.

A further purpose of the meeting was to discuss the extension of PSA to Level 2 and related problem areas for RBMK type reactors such as analysis of accident progression in the core region.

2.2. Activities

The document on "Core damage states for NPPs with RBMK reactors" contains the following presentations which provided the basis for discussions and conclusions during the meeting:

- Framework for PSA Levels 1, 2, and 3, interfaces, and an overview of the present practices in core damage definitions. Accident progression processes and phenomena for vessel type LWRs and channel type reactors including an outline of the US N Reactor accident sequence end states.

- Overview of the Barselina Project PSA. It was pointed out that the main purpose of a PSA is to provide the basis for investigating and implementing plant improvements for design and operational features.

- Fuel and core damage state definitions used in the Barselina Project [5] and an updated version of the approach which includes a first effort to quantify the related releases. The approaches largely follow the one developed for the Darlington PSA in Canada, Ref. [3].

- Processes which might lead to damage to the fuel and the core as well as the margins which exist between the damage assumed based on design limits, and the damage derived from realistic considerations. Technical features and phenomena which are important for the categorization of accidents, criteria for the description of core damage for channel type reactors.

- Present practices and standards used by the regulatory organization for licensing the RBMK NPPs. Requirements regarding consideration of beyond-design events including the development of accident management strategies and procedures.

2.3. Issues and conclusions

2.3.1. Boundary between design basis accidents (DBAs) and beyond-design basis events

The criteria applied for the RBMK reactors to limit situations and events within the design basis are in principle similar to the ones used in other countries for other reactor types. For the topics considered in the discussions, the following criteria are of interest:

Safe operation limits (SOL):

- Limit for the level of radioactivity in the coolant, correlated to a maximum fraction of 1% defective fuel pins which have a gas type leak and a maximum fraction of 0.1% defective fuel pins which have a direct contact of the coolant and fuel.

Maximum design limits (MDL):

- MDL1: Peak cladding temperature should not exceed 1200° C.

- MDL2: Peak cladding oxidation should not exceed 18% of the initial cladding wall thickness.

- MDL3: The fraction of Zirconium reacted should not exceed 1% (weight) of the cladding of fuel pins.

- The energy generated in the fuel for fast power excursions is limited (specific values are given for the energy generated per mass of fuel).

Radiologically, the failure of one fuel channel must be accommodated by the accident localization system (ALS). As a general criterion, all situations and events within the design basis range must fulfill the given offsite dose-limits.

The criteria summarized above can be used to define the boundary for beyond-design events, either by using conservatively the detailed design basis criteria or by carrying out more realistic analyses. Regarding success end states in PSA it should be pointed out here that the success end states in terms of PSA mean that the plant is in a safe and stable end state.

A specific feature of the RBMKs is that the core tank header plate lifts of if more than a few (3 to 9 currently, depends on plant design) fuel channels fail under pressure, see Figure 1 for a schematic picture of the core. There is a relief pipe from the core region to the pressure suppression system, see Figures 2 and 3 showing the two different designs for the ALS with pressure suppression. After core tank header plate lift off, core geometry is lost. However, for many accident sequences this could be prevented by depressurizing the cooling system.

2.3.2. Definition of damage and accident severity

For the purpose of a Level 1 PSA, fuel and core damage for RBMKs can be defined as accident conditions which exceed the design basis limits. This definition of fuel and core damage is essentially the same as the one used for vessel type LWRs; however, it is recognized that for channel type reactors (e.g. RBMKs), "core damage" does not necessarily constitute a gross meltdown of the whole reactor core. The severity or the extent of the core damage is expected to be defined better within PSA Level 2 studies. In the Level 2 studies accident progression, release and transport of radioactive materials within the core, the coolant system, confinement, plant systems and buildings are investigated in detail.

For core damage for the RBMK NPPs the following categories are suggested:

Category P: Beyond design accidents which do not lead to the category G, below;

Category G: Gross core damage which includes unrecoverable loss of core structural integrity. The scenarios which lead to this end states are power excursions, steam separator drum ruptures and lifting of the core upper cover plate.

For beyond design basis accident conditions of Category P it seems appropriate to use "differentiated damage states" to indicate different levels of damage severity and thereby different radiological releases. The definitions of damage states may be plant-specific and developed further as improvements are realized in the deterministic basis.

Since the knowledge base for deterministic assessment of severe accidents in RBMKs and other channel type reactors is fairly limited, deterministic prediction of severe accident progression for RBMK reactors is beyond the current state-of-the-art. Unjustified anticipation of the results of accident progression analysis in the process of defining and developing the damage states should therefore be avoided. The concern in this area is related to underestimation of the extent of damages, for instance by underestimating or omitting processes and phenomena which could cause or expand damage.

An example for the development of differentiated damage states can be found in the N Reactor Level 1 PSA [2]. Intermediate accident sequence end states have been added to the traditional success and complete core damage states of vessel type LWRs. It is pointed out that these states (or bins) represent the fuel initially affected as a result of various system failures and do not necessarily indicate what the final result of the accident will be. According to the report this must be determined by the traditional deterministic thermal-hydraulics and fuel-behavior computer analyses.

The following 5 end states were defined and used in the study:

Bin OK: No fuel damage

Bin W: Failure to deliver ECCS flow to any one of the 16 core inlet risers. All other ECCs functions are successful and the graphite cooling system continues to operate normally, limiting fuel damage within the affected riser.

Bin X: Similar to Bin W, but the graphite cooling system fails. Potential heat removal from adjacent columns of pressure tubes in non-affected risers is conservatively neglected.

Bin Y: Cases where all direct cooling to the fuel (from the primary system and ECCS) is unavailable. Fuel damage is limited due to the heat removal capability of the graphite cooling system.

Bin Z: Cases where 100% of the fuel may be affected due to a complete loss of heat removal capability or a failure to scram. The bin is subdivided into two states Z and Z1 to account for the time course of sequences (Z1 is included to account for delayed fuel damage).

Single tube blockage events were initially examined but not retained as an explicit end state because fuel damage is only expected if less than 1 % of the normal flow cross section area is available and offsite radiological consequences seem to be insignificant.

Thus, the main essence of the additional accident end states for the N Reactor is directly derived from the layout of the cooling system, in particular the inlet risers, which have its analogy in the discharge headers (DH) and group distribution headers (GDH) of RBMKs for example. The procedure for defining differentiated sequence end states would therefore consist in a systematic review of plant systems and their features to identify regional core partitions which are directly affected. The definition would consider initiating events, immediate damage mechanisms and safety related functions, and potential interregional influences.

2.3.3. Available studies and approaches

A few PSA projects have been carried out for NPPs with RBMK reactors. The most comprehensive one is the "Barselina" project in which a Level 1 PSA for the Ignalina NPP is performed in a cooperation project between Lithuania, Russia and Sweden, see Ref. [5].

A limited scope study (called Level 0+) for the Leningrad NPP Unit 1 was carried out by RDIPE, the design institute for RBMK reactors, together with the Kurchatov Institute, see Ref. [4]. An additional limited scope study (Pilot Risk Study, PRS) for the Leningrad NPP Unit 1 was performed in the CEC sponsored project "RBMK Safety Review". This activity has now been extended within the framework of the bilateral cooperation, which RDIPE has with the UK, USA and Sweden.

All these studies have used a similar categorization of accident sequence end states which has been developed following the approach used in the PSA for the Darlington NPP [3].

For the Barselina Project the following core hazard state categories, ordered by increasing consequences, were defined [5]:

Category S, Safe Conditions:

This category applies if:

- the Safe Operation Limits (SOLs) are not exceeded,
- the SOLs are exceeded in no more than 3 channels; the Maximum Design Limits (MDL) are not exceeded in any channel (cladding temperature less than 800° C).

Category V, Violation:

Events fall in this category if:

- SOLs are exceeded in more than three channels due to cladding defects or damages, or
- the MDLs are exceeded in no more than three channels.

Basically no pressure tube ruptures are assumed to occur for Category V. The following subcategories are defined to include few channel tube rupture events:

Subcategory V1, single pressure tube rupture. This category denotes one pressure tube rupture due to channel blockage or another reason. Exceeding of MDLs in one channel is assumed.

Subcategory V2, ruptures of 2 to 3 pressure tubes. This category denotes rupture of 2 to 3 pressure tubes due to channel blockage or other reasons. Exceeding of MDLs in these channels is assumed.

A violation type accident can lead to significant economical consequences in the plant. The consequences for the environment depend on the effectiveness of the accident localization system.

Category D, Reactor Core Damage:

This category is characterized by severe accident conditions caused by significant deviations from design scenarios. It is assumed that MDLs are exceeded in no less than 3 and no more than 90 channels. These conditions may cause rupture of pressure tubes but of no more than three tubes at high pressure (nine pressure tubes after improvement of the reactor cavity steam relief capacity). It is assumed that the temperature of fuel cladding in the damaged channels is higher than 800° C long enough to meet MDL2 or MDL3, or it is higher than 1200° C (MDL1).

Category A, Severe Accident:

This category is characterized by severe accident conditions caused by significant deviations from design scenarios and is accompanied by:

- multiple pressure tube ruptures at high pressure of more than 3 channels (more than 9 channels after improvement of the reactor cavity steam relief capacity),

- exceeding MDLs and fuel damage in more than 90 channels.

One or several events are assumed to happen for this category:

- pressure increase in the reactor cavity due to multiple tube ruptures resulting in lifting the reactor upper plate,
- damage (melting) of fuel of a part of the core or of the whole core,
- fuel dispersion,
- moderator graphite burning.

2.3.4. Further recommendations

The following further recommendations were compiled and discussed during the meeting:

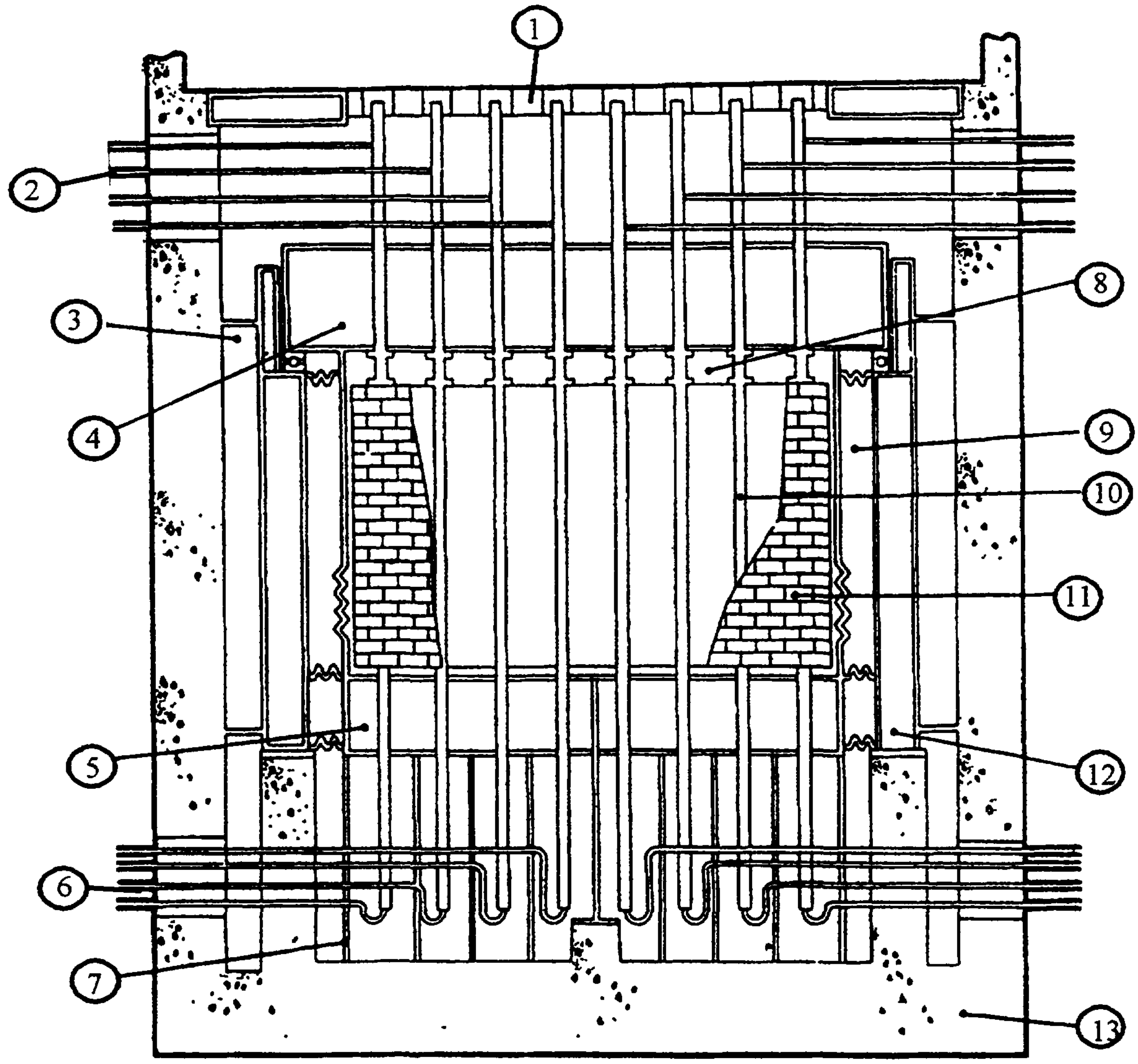
- It has become increasingly evident that detailed deterministic analysis of DBAs and beyond design basis accidents reveal considerable insights into RBMK response to various accident conditions. These analyses can also help in better characterizing the outstanding phenomenological uncertainties, improved EOPs and AM strategies, including potential risk-beneficial accident mitigative backfits.

- It is suggested that the IAEA considers the experience gained from the Barselina Project PSA in anticipation of modifications or extensions of the IAEA PSA procedures.

- An international peer review of the Barselina Project PSA could be beneficial in crossfertilization of experience between various RBMK PSA efforts.

3. SUMMARY

Accident progression and Level 2 analyses are presently not available for RBMK reactors. Accident progression modelling for the core region will require adaptation/extension

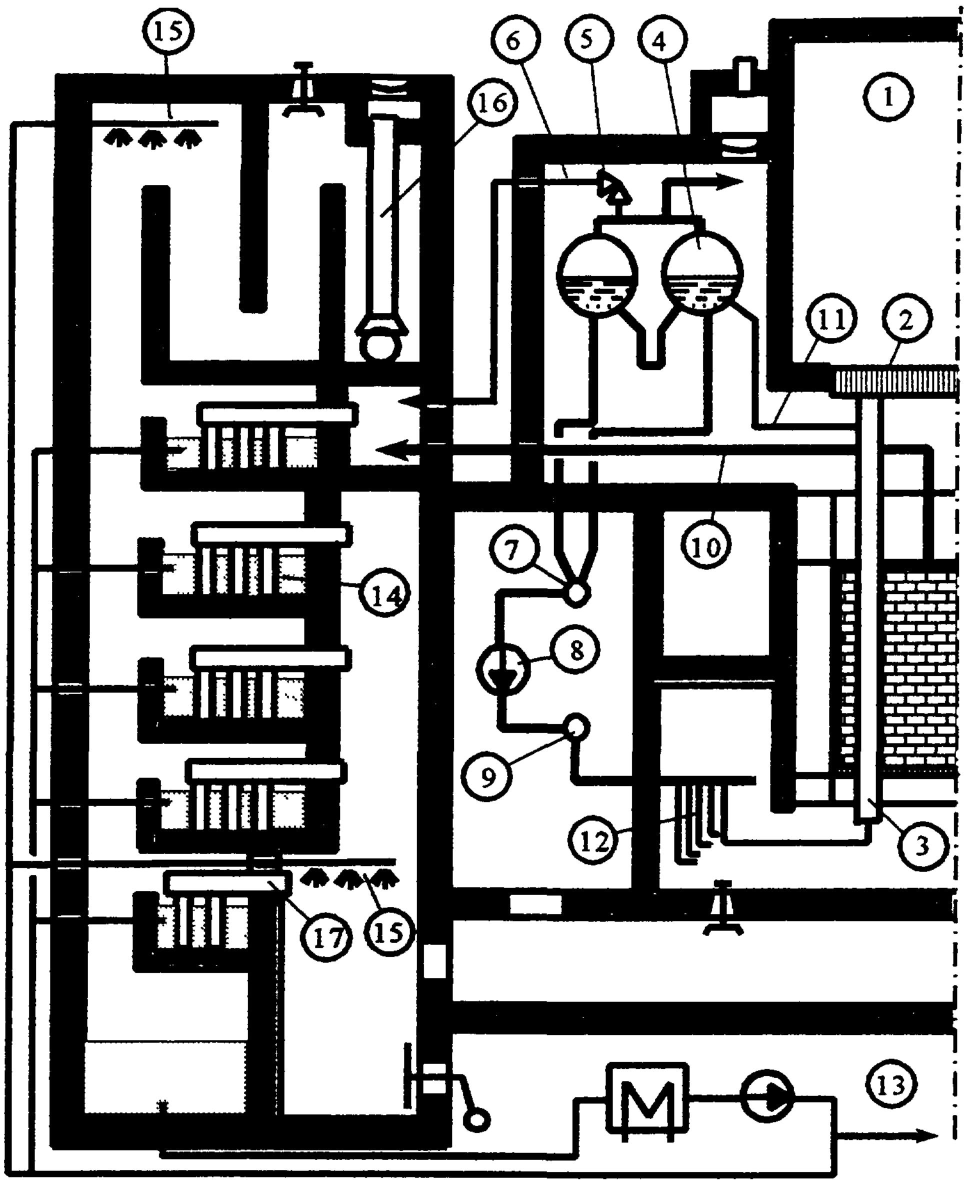


1 Top cover removable floor	8 Helium-Nitrogen
2 Steam water pipes	9 Nitrogen
3 Sand	10 Pressure tube
4 Cover plate	11 Graphite
5 Lower plate	12 Water shield
6 Coolant inlet	13 Concrete shield
7 Support structure	

FIG. 1. Reactor Core

of existing computer codes. Experimental background regarding modelling of accident progression in the core region might be limited. Such kind of analysis might be simpler than for other channel reactors, as

- Fuel channels are vertical and quite decoupled from each other,
- Moderator material and structure looks very stable, channel tube - moderator configuration is relatively simple,
- Fuel channel failure propagation modelling does not appear too difficult.



1 Central hall	7 Suction header	13 Spray system
2 Top cover	8 MCP	14 Condensation pools
3 Fuel channel	9 Pressure header	15 Spray header
4 Steam separation drum	10 Discharge pipe	16 Discharge pipe section
5 Safety/relief valve	11 Steam water pipes	17 Steam discharge pipe
6 Discharge pipe from SRV	12 Lower water pipes	

FIG. 2. Accident Localization System

Determination of the capability of the ALS (Accident Localization System) system to accommodate severe accident thermohydraulic and radiological loads, taking into account bypass paths through leakage from confinement compartments to the atmosphere is recommended. Significant backfitting of existing confinements are planned. This backfitting should include consideration of PSA level 1 results and analysis of severe accidents. For this kind of analysis computer codes are available and applicable.

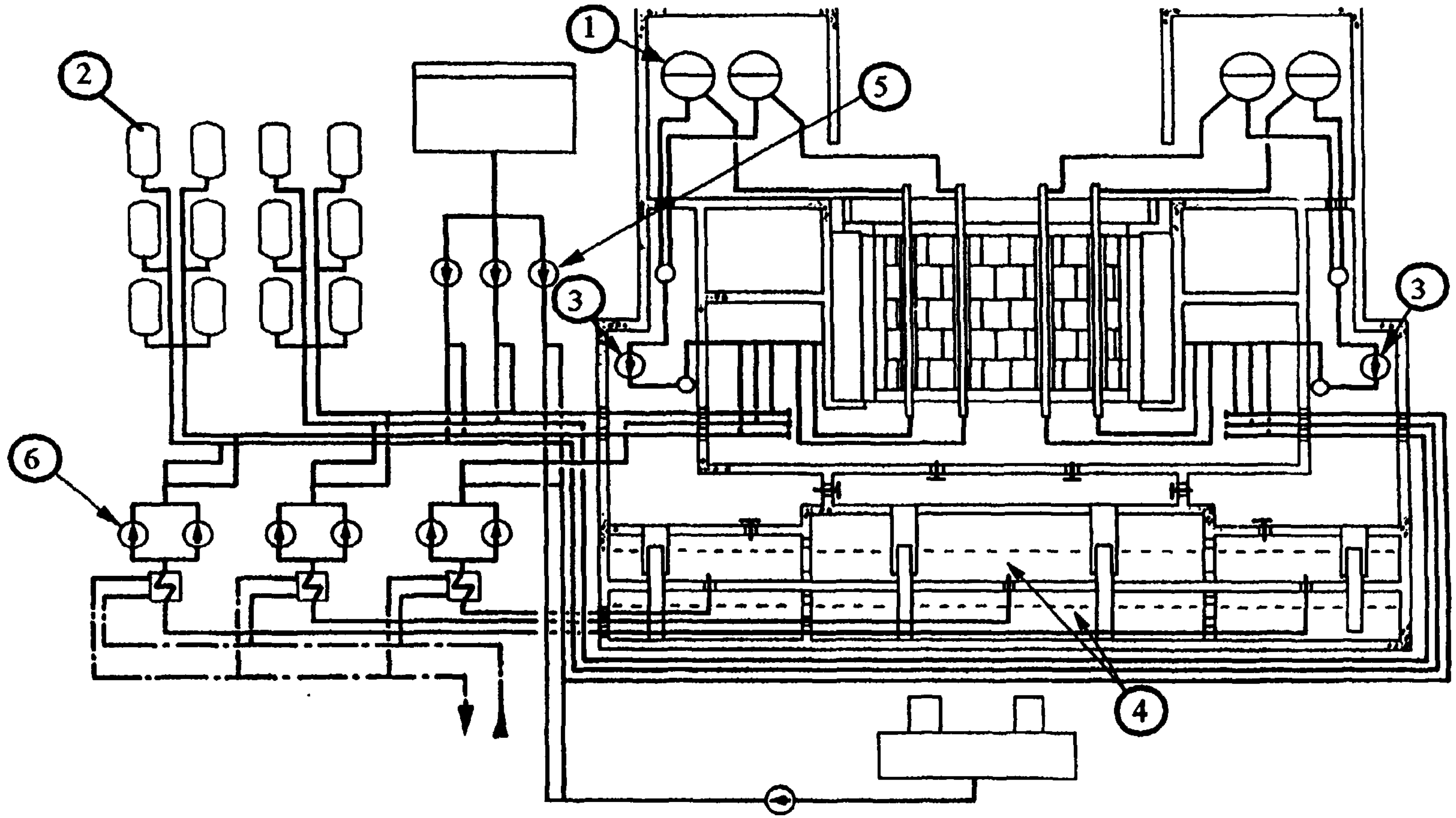


FIG.3. Accident Localization System

1 Steam separator drums
2 Accumulators
3 MCP
4 Suppression pools
5 ECCS pumps
6 ECCS pumps

ABBREVIATIONS

ACS	accident confinement system
ALS	accident localization system
AM	accident management
DBA	design basis accident
DH	discharge header
ECCS	emergency core cooling system
EOP	emergency operating procedure
GDH	group distribution header
INSAG	International Nuclear Safety Advisory Group
LWR	light water reactor
MCP	main coolant pump
MDL	maximum design limit
PSA	probabilistic safety assessment
PSC	probabilistic safety criteria
RBMK	light water cooled, graphite moderated, channel type reactor (Soviet design)
SOL	safe operation limits
SRV	safety/relief valve

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